GA–A23598

DIII–D THREE-YEAR PROGRAM PLAN 2001–2003

by
R.D. STAMBAUGH and RESEARCH STAFF OF THE DIII–D TEAM

JANUARY 2001

DISCLAIMER

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ABSTRACT

This three year program plan presents a summary of the research planned on the DIII–D tokamak in the years 2001–2003. Reference is made to GA–A22950, "The DIII–D Five-Year Program Plan," which is a comprehensive discussion of research planned for DIII–D in the period 1999–2003.

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1. EXECUTIVE SUMMARY

1.1. MISSION OF THE DIII–D NATIONAL FUSION PROGRAM

The overall mission statement of the DIII–D Program is “To establish the scientific basis for the optimization of the tokamak approach to fusion energy production.”

The main output of the DIII–D Research Program is a scientific basis. “Scientific” means developing a solid understanding of the underlying physical principles and incorporating them into useful predictive modeling tools. “Optimization” means experimentally demonstrating performance parameters at or near the theoretically predicted limits for the tokamak magnetic confinement system and achieving to the greatest degree possible an integrated, steady-state demonstration of optimized performance that projects to an attractive fusion power system. The integrated optimization sought and the scientific basis established will allow the definition of optimal paths to fusion energy using the tokamak approach.

1.2. DIII–D NATIONAL PROGRAM RESEARCH GOALS

Working with our international colleagues, the DIII–D Program pursues the above mission through three additional research goal statements.

1. The DIII–D Program's primary focus is the Advanced Tokamak (AT) Thrust that seeks to find the ultimate potential of the tokamak as a magnetic confinement system.
2. Where it has unique capabilities, the DIII–D Program will undertake the resolution of key enabling issues for advancing various magnetic fusion concepts.
3. The DIII–D Program will advance the science of magnetic confinement on a broad front, utilizing its extensive facility and national team research capability.

The DIII–D National Research staff is highly motivated to pursue the Advanced Tokamak (AT) Thrust. Finding the ultimate potential of the tokamak as a confinement system is primarily a scientific motivation and a complex scientific endeavor. The optimization and integration of advanced tokamak elements into achievable single discharges requires tradeoffs evolved over a multi-year period.

Research on DIII–D toward these goals is organized in a matrix type of approach in which one dimension of the matrix is a set of thrusts (Fig. 1). A thrust is aimed at a key scientific objective and is given a significant block of run time. Thrusts provide a more purposeful and visible path to pursuing critical research subjects. The complexity of some research subjects require broader multi-disciplinary teams and the research thrusts provide a way to nucleate such teams. Most but not all of the thrusts relate to key elements of the AT goal of the DIII–D Program. Eventually it is necessary to integrate various key scientific investigations into single discharge scenarios, which in general require compromise and trade-offs among physics constraints. Thrusts also provide the means to focus efforts from various scientific lines into these integrated discharge scenarios. Pursuit of these scenarios also helps define the key issues that need to be addressed in the other thrusts and topical science areas. The research thrusts and their leaders will change year-to-year to keep up with the evolution of the experimental program.

The second dimension of the experimental planning matrix is comprised of the four enduring topical areas of fusion energy science: stability, confinement and transport, divertor/edge physics, and heating and current drive. The DIII–D Facility and the DIII–D National Team are resources of immense value to the U.S. Fusion Program in terms of advancing the science of magnetic confinement on a broad front. DIII–D has a superb

2001 Research Thrusts and Leaders

	#1 Edge Pedestal	#2 High Bootstrap Fraction AT Scenario	#3 Neoclassical Tearing Modes	#4 Wall Stabilization	#7 Internal Transport Barriers
Topical Area Manager	R. Groebner T. Osborne	M. Wade T. Luce J. Ferron	R. La Haye C. Petty	A. Garofalo L. Johnson	E. Doyle C. Greenfield
Stability physics E. Strait	✓	✓	✓	✓	✓
Confinement, transport physics K. Burrell	✓	✓			✓
Divertor, edge physics S. Allen	✓	✓			✓
Heating and current drive physics R. Prater	✓	✓	✓		✓

Fig. 1. The DIII-D Research Planning Matrix.

diagnostic set, increasingly flexible and capable plasma control systems, an excellent research staff, and a comprehensive set of analysis codes and theory support that enable real learning in depth from the experiments done. The staff recognizes and embraces a responsibility to the greatest extent possible to use that resource to advance the state of fusion energy science knowledge generally. The managers of these topical areas implement this second dimension of the matrix Their continuing leadership of these topical areas over a period of years assures the continued scientific focus of the DIII–D research.

The DIII–D Research Staff also are strongly motivated to see magnetic confinement progress to future next-step devices. The AT work and the broader scientific work on DIII–D can contribute greatly to the definition and the support for these future machine initiatives. Some of those possible next step options are:

An international D-T burning plasma experiment such as the ITER-FEAT which plans more exploitation of and/or reliance on AT physics.

An advanced performance superconducting tokamak (JT–60SC, ARIES-AT) which exploits AT physics toward steady-state.

A copper-coil ignition experiment about the size of JET and using gyroBohm scaling of H–mode, relying on more conventional tokamak physics.

A compact, high magnetic field copper-coil ignition experiment (as exemplified by CIT/BPX/IGNITOR) but enabling studies of or relying on AT physics (FIRE).

A next-step spherical torus which relies on most elements of AT physics to enable the study of burning plasma physics in long pulse or steady-state.

Research toward Goal 2 (key enabling issues) can appear either as thrusts or as elements of the Topical Science Area plans. A discussion of how DIII–D research relates to the various future machine possibilities can be found in Section 2.4 of the Five-Year Plan.

Competition for experimental time on DIII–D is intense. Priority goes to the Advanced Tokamak work, which occupies most of the thrusts. We seek to reserve about 30%–40% of the run time for the Topical Science Area Managers to allocate within their discretion to more broadly motivated studies. The work to support Goal 2 has to find time either as a thrust or in the Topical Areas.

1.3. DIII–D AND FESAC FIVE-YEAR GOALS

In this section, we outline how in the next five years the DIII–D National Program will make major contributions to the newly defined FESAC Goals and Objectives for Magnetic Fusion Energy (MFE) (see Table 1 Goals and Near-Term/Long-Term

Objectives for MFE, “Report of the FESAC Panel on Priorities and Balance,” September 1999).

1.3.1. FESAC GOAL 1 (from the IPPA Report)

Goal: • Advance scientific understanding of turbulent transport, forming the basis for a reliable predictive capability in externally controlled systems.

In the area of high temperature plasma science the DIII–D combines an internationally unexcelled capability for reproducibly producing high temperature plasmas in a wide range of plasma shapes, a unique ensemble of plasma diagnostics with outstanding spatial and temporal resolution, and close coupling to the exceptional U.S. MFE theory and modeling community. The FESAC 5-year objectives and specific DIII–D 5-year research directions are:

- **Turbulence and Transport:** *Advance understanding of turbulent transport to the level where theoretical predictions are viewed as more reliable than empirical scaling in the best understood systems*.

  An overall goal for the DIII–D program is to work towards a predictive understanding of tokamak transport. Achieving this goal requires the combined efforts of theorists, modelers, and experimentalists to develop the fundamental theories, include them in numerical models, compare those models with the results of experiments and then iteratively improve them. The key issues here are understanding turbulent transport in both the electron and ion channels. Our work over the next few years will include fundamental investigations of the nature of tokamak turbulence, comparison of those turbulence measurements with predictions of gyrokinetic and gyrofluid codes, and definitive tests of present-day transport models in well-diagnosed plasmas using both steady-state and modulated techniques. In addition, we will be further testing the model of E×B shear suppression of turbulence by utilizing various techniques (e.g. impurity injection, electron heating) to investigate the functional dependence of the turbulence growth rates on these plasma parameters. Finally, by use of E×B shear to stabilize the longer wavelength, ion temperature gradient modes, we will attempt to isolate and investigate the shorter wavelength modes which primarily affect electron transport.

- **Macroscopic Stability:** *Develop detailed predictive capability for macroscopic stability, including resistive and kinetic effects*.

  DIII–D is conducting experiments aimed at validating theoretical models for ideal, resistive, and kinetic plasma instabilities using experimental measurements adequate for quantitative tests of the theoretical calculations. A principal focus is on the resistive wall mode, the manifestation of the low order kink mode in the presence of a conducting wall, which sets the two most important operation space limits for the tokamak, the pressure and current limits. The goal is to extend the DIII–D performance to the theoretical limits of stability and to develop the intellectual, computational, and laboratory tools necessary to apply these results to other devices.

- **Wave-Particle Interactions:** *Develop predictive capability for plasma heating, flow and current drive, as well as energetic particle driven instabilities, in power-plant relevant regimes.*

  The DIII–D Program will develop methods of plasma current generation (initiation, rampup, sustainment, and profile control) to provide future devices the basis for full steady-state transformerless operation. DIII–D is developing the physics basis being embodied in predictive codes for electron cyclotron, fast wave, and neutral beam current drive and for maximal use of the self-driven bootstrap current. Electron cyclotron waves are used for heating, fundamental transport studies with pulse modulation techniques, current drive, local pressure and current profile control, and stabilization of tearing modes.

- **Multi-Phase Interfaces:** *Advance the capability to predict detailed multi-phase plasma-wall interfaces at very high power- and particle- fluxes.*

  DIII–D will bring 2-D measurements of divertor plasma properties into comparison with 2-D predictive code calculations of those properties. The DIII–D principal research direction will be to maximize the degree of recombination and radiation in the divertor plasma in order to minimize heat fluxes to and erosion of plasma facing surfaces. Detached plasma states with high recombination fractions have been found and successfully simulated in the 2-D codes. The frontier task is to achieve these regimes in the lower density plasmas optimal for current drive and high bootstrap fractions.

1.3.2. FESAC GOAL 2

Goal: • *Resolve outstanding scientific issues and establish reduced-cost paths to more attractive fusion energy systems by investigating a broad range of innovative magnetic confinement configurations*.

The Advanced Tokamak vision of the ultimate potential of the tokamak as defined by theory work has extremely hollow current profiles and nearly 100% self-organized bootstrap current produced by high quality transport barriers near the plasma edge. These equilibria are certainly highly innovative and are so different from normal tokamak experience as to essentially constitute an alternate concept. Studies have shown that these modes, if realized, can halve the cost of electricity in tokamak fusion power systems.

The experimental and theoretical research DIII–D carries out in pursuit of the AT vision has many elements of generic value across magnetic confinement concepts:

- **Electrostatic Turbulence Suppression:** The mechanism of stabilization turbulence by sheared E×B flows, pioneered by DIII–D, appears to be universal across magnetic confinement concepts and is a continuing focus of DIII–D research.

- **Wall Stabilization:** The physics and technology of stabilization of modes by a nearby conducting wall and feedback coil system being investigated on DIII–D is a development necessary for the advanced tokamak, spherical torus, RFP, spheromak, and FRC.

- **Energetic Particle Density Gradient Driven Instabilities:** The study of these instabilities was identified as having generic value across concepts at the Snowmass Summer Study. Such modes, excited by the fast ions from the neutral beams, are an important subject of study in DIII–D.

- **Current Drive by Waves and Beams:** The wave-particle and beam-plasma interaction physics, developed in the tokamak generally and DIII–D in particular, for driving current is largely generic across magnetic concepts.

- **Parallel Field Line Physics:** The physics investigations in the scrape-off layer and divertor plasmas is largely generic across concepts because of the dominant role of the parallel heat and particle flows and the prominence of concept non-specific atomic physics. The DIII–D divertor research emphasis on systematic experiments, 2-D diagnostic measurements, and modeling enables transfer to other concepts of the understanding gained.

1.3.3. FESAC GOAL 3

Goal: • *Advance understanding and innovation in high-performance plasmas, optimizing for projected power-plant requirements; and participate in a burning plasma experiment.*

The DIII–D National Program was instrumental in defining the Advanced Tokamak concept. This vision of the ultimate potential of the tokamak as defined by theory work has extremely hollow current profiles and nearly 100% self-driven bootstrap current produced by high quality transport barriers near the plasma edge. Theory predicts that with wall stabilization of ideal modes the beta limit in the tokamak can be about twice the free boundary limit. Transport rates as low as neoclassical in the ions are envisioned and have been seen in experiments. Detached, highly recombining divertor operation needs to be combined with these advanced core plasma modes. Studies have shown that these modes, if realized, can halve the cost of electricity in tokamak fusion power systems and enable modest sized burning plasma experiments reaching for high gain and steady-state.

The FESAC 5-year objectives and specific DIII–D 5-year research directions are:

- *Assess profile control methods for efficient current sustainment and confinement enhancement in the Advanced Tokamak, consistent with efficient divertor operation, for pulse length >> τ_E.*

  Efficient current sustainment will be achieved on DIII–D by maximizing the bootstrap current and supplementing that with electron cyclotron, fast wave, and neutral beam current drive. Near term scenarios being pursued aim at bootstrap fractions over 50% and sustained with current profile control (for up to 5 seconds) by microwave electron cyclotron heating (ECH) power in a divertor plasma with a normalized beta of 4 and an energy confinement enhancement 2.5 times L–mode. Parallel lines of research on transport barrier physics and divertor physics in closed, pumped divertors are laying the groundwork for eventual long pulse integrated scenarios beyond the near term work.

- *Develop and assess high-beta instability feedback control methods and disruption control/amelioration in the Advanced Tokamak, for pulse length >> τ_E.*

  DIII–D is developing the physics and the technology of stabilization of kink modes by a conducting wall backed by non-axisymmetric feedback coils. Stabilization of neoclassical tearing modes by direct application of electron cyclotron current drive (ECCD) and indirect methods of current profile alteration by ECCD will be more thoroughly explored. DIII–D has an extensive program of stability studies and plasma control development aimed at enabling disruption free operation close to stability limits. The injection of impurity pellets, massive gas puffs, or liquid jets shows promise of success at providing a means of ameliorating the consequences of disruptions.

1.3.4. FESAC GOAL 4

Goal: • *Develop enabling technologies to advance fusion science; pursue innovative technologies and materials to improve the vision for fusion energy; and apply systems analysis to optimize fusion development.*

The DIII–D will deploy, and thereby foster, the development of a number of enabling and innovative technologies. Most notable are advanced methods for plasma heating and current drive [microwave electron cyclotron radio frequency (ECRF)]; disruption mitigation by solid, liquid, or gas injection; plasma fueling (inside pellet launch); plasma flow control [neutral beam, ECRF, ion cyclotron radio frequency (ICRF)]; investigation of novel divertor concepts; feedback technologies for wall stabilization; studies of surface erosion; and small-sample testing of low activation materials in plasma environment.

1.3.5. SUMMARY

Within world fusion science research, the DIII–D National Program aims to retain leadership in advanced tokamak research and in high temperature plasma science. In so doing, results from DIII–D research will be of benefit to other magnetic confinement configurations and will serve as a test bed for several enabling and innovative technologies.

1.4. THE DIII–D PROGRAM — INTERNATIONAL CONTEXT

DIII–D advanced tokamak research is carried out with extensive international collaboration to provide opportunities for scientific confirmation and joint experiments. Worldwide tokamaks (with characteristics listed in Table l) have research programs which differ and complement each other; a summary of research capabilities is given in Table 2. International databases enable documenting accomplishments, comparing results of experiments and theory, and coordinating research. U.S. tokamaks make vital contributions to the world program with a focus on concept innovation and optimization.

Table 1
Characteristics of Operating World Tokamaks

	Plasma Current (MA)	Magnetic Field B(T)	Major Radius R (m)	Comment
Performance Extension Tokamaks				
JET	6.0	4.0	3.0	E.U.
JT–60U	3.0	4.4	3.3	Japan
DIII–D	3.0	2.1	1.7	U.S.
Alcator C–Mod	2.0	9.0	0.65	U.S.
Tore Supra	1.7	4.0	2.3	France (superconducting)
ASDEX Upgrade	1.6	3.1	1.7	Germany
Proof-of-Principle Tokamaks				
FT–U	1.6	8.0	0.93	Italy
TCV	1.2	1.4	0.88	Switzerland
TEXTOR	1.0	3.0	1.75	Germany
JFT–2M	0.5	2.2	1.3	Japan
T–10	0.4	3.0	1.5	Russia
Compass-D	0.4	2.1	0.55	England
Triam-1M	0.15	8.0	0.84	Japan (superconducting)
Concept Exploration Tokamaks (partial list)				
JFT–2M	0.5	2.2	1.3	Japan
ET	0.3	0.25	5.0	U.S./UCLA
Truman-3M	0.18	1.2	0.5	Russia
HBT–EP	0.025	0.35	0.95	U.S./Columbia U.
Steady State Tokamaks (under construction)				
KSTAR	2.0	3.5	1.8	Korea (2004)
HT–7U	1.0	3.5	1.7	China (2004)
SST–1	0.22	3.0	1.1	India (2002)

Table 2
World Advanced Tokamak Research Capabilities

Research Facility	Unique Research Capability
Performance Extension Tokamaks	
JET (E.U.)	DT capability at large size
JT–60U (Japan)	Long pulse high performance physics at large size
DIII–D (GA)	AT physics, high shape flexibility, high beta, divertor, ECH
Alcator C–Mod (MIT)	High field, high density divertor
Tore Supra (France)	Long pulse superconducting
ASDEX Upgrade (Germany)	Divertor physics, ECH, advanced issues
Proof-of-Principle Tokamaks	
FT–U (Italy)	High field, IBW
TCV (Switzerland)	High elongation
Concept Exploration Tokamaks	
ET (UCLA)	High beta via omnigeneity
HBT–EP (Columbia U.)	High beta via feedback

Comparing results from DIII–D with the two larger higher-temperature European and Japanese devices provides an opportunity to extend DIII–D research results and understanding to a larger scale. The European JET can operate with D-T plasmas, while the Japanese JT–60U research focuses on steady-state high-performance plasmas. Three mid-size divertor tokamaks are equipped with sufficient plasma heating, control, and diagnostic systems to carry out advanced tokamak research on a broad front. DIII–D is a low-field tokamak with high power heating including ECH for high-beta advanced tokamak research. DIII–D is unique worldwide with its poloidal field magnet capability for extensive research in plasma shaping and to emulate other tokamak shapes for coordinated joint research. Alcator C–Mod is the world's highest-field tokamak, capable of very high-density operation with equal electron and ion temperatures, with plasma pressure equal to that expected in a reactor. Its compact size and closed divertor configuration offer unique capabilities for studying high power-density plasma exhaust physics. Together DIII–D and Alcator C–Mod provide data from two plasmas with very different physical parameters but similar dimensionless parameters. The German ASDEX-Upgrade has external plasma shaping control coils of more reactor relevance but with less shape flexibility than DIII–D. Three non-divertor tokamaks, TEXTOR, FTU, and Tore Supra address pumped limiter, high field physics and steady-state current drive, and heat removal respectively. Korea is constructing a superconducting advanced tokamak (KSTAR), and China is engineering the design of a superconducting tokamak (HT-7U). Japan is considering a superconducting device JT-60SC to replace JT-60U. DIII–D collaborates with all these international tokamaks.

Two U.S. experiments contribute to tokamak concept exploration. The Columbia University high beta tokamak (HBT-EP) is addressing wall stabilization and active mode control, issues critical for advanced tokamak operation now being extended to DIII–D. The University of California Los Angeles (UCLA) Electric Tokamak (ET) is a low-curvature electric tokamak built to explore the possibility of achieving classical confinement and unity beta in tokamaks.

The National Academy has suggested research program strengths can be classified as to their leadership uniqueness in the context of related world programs. In this respect, DIII–D is unique in its plasma shape flexibility, its high beta research including feedback stabilization, its comprehensive transport diagnostics, its ECCD profile control capability, and its advanced tokamak divertor program. DIII–D pioneered advanced tokamak concepts through an integrated approach to fusion energy science and is a leading supplier of results to international physics databases. DIII–D is among world leaders in ICRF (having pioneered fast wave current drive), in the study of neoclassical tearing modes (collaborating with AUG and JET), and in pellet fueling. DIII–D does not commit significant resources to a number of research areas where others have strong leads. These areas include large scale facility size, D-T capability, LHCD, and metallic divertor. From the above classification it is evident that DIII–D strives for leadership in several areas of fusion science and physics innovation rather than in fusion technology where other world facilities lead.

1.5 WHAT ARE THE KEY SCIENTIFIC QUESTIONS BEING ADDRESSED IN THE DIII–D PROGRAM?

As is clear from the discussion above, the DIII–D Program is a very broad program and pursues many scientific questions. Boiling the program down to a short list of questions is difficult. Comprehensive lists of research goals made by the Thrust and Topical Area Leaders can be found in Sections 3 and 4. Here we attempt what might be called the program director's view of the critical research questions. In order to provide background and context for the critical questions, it is useful to list in order what we believe we do know, what we think is true but have not quite finished the scientific proof, and finally what the key remaining open questions are.

1.5.1 WHAT WE BELIEVE HAS BEEN ESTABLISHED

Stability

The principal operating space limits for the tokamak, the pressure and current limits, are well described by thresholds for ideal kink mode stability. Crossing these limits is what causes disruptions.

In the presence of a conducting wall, the limiting ideal kink mode manifests itself as the resistive wall mode (RWM). This mode does exist and has been observed.

Neoclassical tearing modes (NTMs) can be stabilized by localized electron cyclotron current drive.

Heating and Current Drive

Plasma current can be driven by electron cyclotron and fast waves and neutral beams.

High non-inductive current fractions can be achieved.

Steady-state requires bootstrap fractions over 50%. Stable equilibria with bootstrap fractions over 50% have been calculated and require wall stabilization so that β_N can be greater than 4, which also requires some confinement enhancement.

Confinement

Sheared E×B flows can suppress or even completely stabilize turbulence in the ion gyroradius wavelength range, forming both the edge H–mode and the internal transport barriers.

Ion transport as low as the theoretical minimum neoclassical transport can be achieved.

Boundary Physics

Suitably radiating divertors can be achieved at high density with recombining divertor plasmas (predicted by the codes) and zero net surface erosion.

Divertor erosion in attached plasmas is consistent with model calculations.

Densities above the Greenwald limit with good confinement can be achieved.

Multi-Disciplinary Issues

States with high stability, confinement, and bootstrap fraction around 50% can be achieved for modest durations.

Long duration plasmas in apparent steady-state can be achieved with stability, confinement, and bootstrap fractions above conventional tokamak behavior.

1.5.2. WHAT WE BELIEVE TO BE THE CASE, BUT STILL REQUIRE FURTHER SCIENTIFIC INVESTIGATIONS

Stability

Non-axisymmetric coil feedback systems can affect the threshold and growth of RWMs.

The physics of the neoclassical tearing mode is described by current theory.

The edge pressure gradient is limited by intermediate n kinks [edge localized modes (ELMs)]; the edge bootstrap current is calculated to play a key role in the stability of the edge.

Disruptions occur when major stability limits are crossed. It is possible to operate disruption free close to major stability limits with sufficient plasma control.

The consequences of disruptions can be successfully mitigated by massive gas puffs or a liquid jet.

Alfvén eigenmodes in certain regimes have an important influence on fast ion confinement.

Heating and Current Drive

Electron cyclotron current drive efficiency and spatial localization are in agreement with theory.

Confinement

The ion temperature gradient mode is mainly responsible for ion transport.

Boundary Physics

The physics of the density limit arises mainly from power balance considerations in the scrape-off layer (SOL) and divertor, tearing mode stability, and stiff transport profiles in the plasma interior. In beam heated H–mode discharges, the plasma within the closed flux surfaces is thermally stable.

Impurity enrichment in the divertor can be produced by scrape-off layer plasma flows.

E×B drifts around the X-point are important, for example in 2-D plasma flows and recycling patterns and the power threshold for the L-H transition.

Multi-Disciplinary Issues

Wall stabilization can enable weak or negative central shear states with by $\beta_N \sim 4$ and long duration.

The operating space for the quiescent double barrier mode is interestingly large.

1.5.3. WHAT ARE THE MAJOR REMAINING OPEN QUESTIONS?

Stability

The free boundary stability limit on pressure is given approximately by $\beta_N \sim 2.8$. With wall stabilization, the tokamak operating space is theoretically predicted to be about twice the free boundary limit. Can wall stabilization enable operation in this higher beta regime and what are the dimensions of that regime?

Is it possible to avoid neoclassical tearing modes, rather than stabilizing them?

Is the edge bootstrap current really there and in agreement with theory?

What physics determines the size and spatial extent of ELMs?

What edge instability is responsible for quiescent H–mode behavior?

How can we account for disruptions away from the pressure and current limits?

Heating and Current Drive

Can ECCD be used to make the hollow current profiles consistent with stability for steady-state?

Can fast waves provide sawtooth stabilization, the key to the high internal inductance AT scenario?

Confinement

What is the physics that triggers the L-H transition?

Can we move transport barriers to large radius and control its pressure gradient for stability?

What modes determine electron transport? [Electron temperature gradients (ETG) are a candidate.]

What physics allows transport barrier formation in the electron channel. (α-stabilization is a candidate.)

Is turbulent transport derived from continuous or intermittent process (avalanches)?

Do zonal flows exist and what role do they play in transport and transport barriers?

Can we develop models of particle and momentum transport?

Boundary Physics

Can suitably radiating divertor states be realized at low density?

By what transport paths are plasmas refueled?

What is the source and path for carbon to get into the core plasma?

Can good confinement at high density be extended to higher betas and/or other geometries?

Multi-Disciplinary Issues

What sets the width of the H–mode edge pedestal (and hence the height given a magnetohydrodynamic (MHD) limit on the edge pressure gradient)?

Can an operating scenario with full non-inductive current drive and over 50% bootstrap fraction, *weak to negative central shear*, enhanced confinement, high beta with wall stabilization, and an adequately radiating divertor be achieved at suitably low density?

Can we realize a *high internal inductance* scenario with slightly over 50% bootstrap current, full non-inductive current drive, enhanced confinement, adequate stability with free boundary plasmas, and an adequately radiating divertor be achieved at suitably low density?

Can the internal transport barrier research, building on *the quiescent double barrier mode*, lead to a scenario with full non-inductive current drive, enhanced confinement, high beta with wall stabilization, and an adequately radiating divertor at suitably low density?

1.6. RESEARCH PLAN LOGIC

Our long range AT Program presently envisions a gradual evolution toward finding out what the ultimate potential of the tokamak can be as a magnetic confinement configuration. In many cases, we are developing as separate thrusts the scientific basis of the building blocks of Advanced Tokamak physics: wall stabilization, neoclassical tearing mode physics, edge plasma stability, transport barrier physics, plasma shape optimization, density and impurity control. In parallel, as the maturity of the building block subjects allows, we seek to integrate these AT elements into single discharge scenarios that sustain high performance for long times. Two intermediate scenarios, are described in Sections 2.2.1 and 2.2.2 and will be carried out with lower toroidal field, plasma current, and electron current (EC) power less than our ultimate objective. Achievement of these intermediate objectives will provide a basis point for pressing on to more adventurous scenarios that probe the ultimate potential of the tokamak. That potential, as defined by theory calculations of stability and the residual transport after ion temperature gradient (ITG) turbulence is suppressed, involves very broad pressure profiles, transport barriers near the plasma edge, nearly 100% bootstrap current in a peak near the edge, and very high normalized beta supported by effective wall stabilization systems. These more challenging investigations as well as our intermediate scenarios will be extended to 10 second pulses at full (2.1 T) toroidal field in DIII–D. These ultimate scenarios are described in Section 2.2.3.

1.6.1. HIGH BOOTSTAP FRACTION SCENARIO (1999 Progress)

The logic diagram for the DIII–D Advanced Tokamak Program has been revised to take account of progress in 2000 and is shown in Fig. 2. The main line is presently seen as the pursuit of the high bootstrap fraction AT scenario. This scenario is described as the negative central shear scenario in Section 2 because it derived from the exciting negative or reversed shear discoveries in tokamaks in the last few years. However, since our results to date have been most positive with weak or slightly positive shear, we have

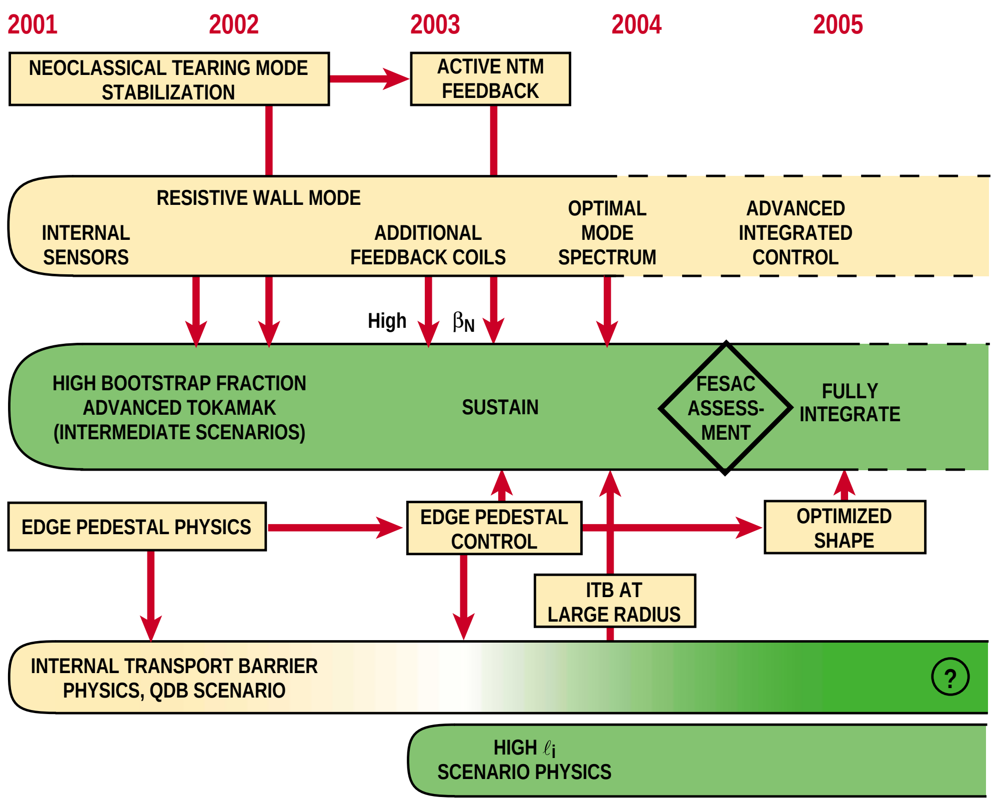

Fig. 2. Logic diagram for the DIII–D Advanced Tokamak Program.

called this line of AT research simply the high bootstrap fraction scenario in this diagram and in our research thrust lists. The matter of whether the optimal shear is negative or weakly positive will be decided by the research. In 1999. we established the basic feasibility of this scenario to deliver high performance plasmas for modest durations. Without any active control of the current profile or the density, discharges with β_N H_{89} ~ 9 for 2 s (16 energy confinement times) were obtained with β_N = 3.8 and β_T = 4.5%. Analysis of the internal loop voltage in this type of discharge indicated that about 75% of the plasma current was supplied non-inductively with a bootstrap fraction of 50%. The duration of these discharges was limited by the uncontrolled inward diffusion of the current profile which resulted in the growth of a resistive wall mode. These plasmas and this strategy at this time essentially sought to produce an integrated AT demonstration within the constraints of free-boundary stable plasmas.

1.6.2. HIGH BOOTSTAP FRACTION SCENARIO (2000 Progress)

Anticipating the need for off-axis ECCD to counteract the current diffusion, we completed for the 2000 campaign the upper divertor to allow density control in high triangularity (δ ~ 0.7) upper single null plasmas so that the necessary ECCD efficiency could be obtained. This divertor was successful in producing the density control needed to enable sufficient ECCD efficiency for our scenario. Also a reduction of low Z (carbon) impurities by about a factor of two was an added benefit. However, since δ ~ 0.9 in the originally developed scenario, this installation required the use of a plasma shape somewhat different from the optimized shape used in 1999. Achievable values of β_N for longish durations in 2000 were 10%–15% smaller with a shape optimized for pumping than values achieved in the optimized shapes of 1999. We are still working on whether this apparent shape dependence of the beta limit can be obtained from theory calculations. In 2000, since the divertor was available to provide the density control previously lacking, we sought to find out how long a duration discharge we could make at high performance. The result was a discharge that lasted 6.3 seconds, (35 energy confinement times) with a product β_N*H_{89P} = 7.5 with an ELMing H–mode edge. The lower β_N in these plasmas also meant lower bootstrap fraction f_{BS} ~ 40%. Beta was regulated constant about 5%–10% below the 2/1 tearing mode limit (these discharges had a stationary 3/2 tearing mode present). The density was also regulated constant by gas fueling and divertor pumping. These discharges showed the kind of long pulse integrated performance we are seeking to demonstrate, but we desire eventually higher β_N, H factor, and bootstrap fraction.

1.63. WALL STABILIZATION

While it has always been recognized the wall stabilization is the key to opening that large portion of the tokamak operating space that lies above the free boundary beta limit, it is clear for this AT line that wall stabilization is on the critical path to higher β_N. The wall stabilization research thrust needs to continue its scientific development as a separate activity this year. New internal sensors have been installed and must be incorporated in the feedback methodologies. The RWM thrust in 2001 will seek to demonstrate a long duration sustainment of a plasma clearly above the no-wall limit. There may be late in the campaign an attempt to apply the RWM feedback to the benefit of the High Bootstrap Fraction scenario, but realistically this work is probably best left for the 2002 campaign. For the 2003 campaign, we plan to have expanded the feedback coil set to 18 coils from the present 6 coils. The more optimal mode spectrum that results is predicted to make possible 80% of the theoretical gain from the no-wall limit to the ideal wall case. Success in this line will be the key to opening path to AT operation.

1.6.4. ELECTRON CYCLOTRON CURRENT DRIVE

With the density control needed for ECCD established, the next order of business in this research line is to apply ECCD power to counteract the resistive diffusion of the current. Gradual growth in EC power and pulse length in the period 2001–2003 will enable longer pulse sustainment of the AT scenarios. As more ECCD power becomes available throughout 2001–2003, we will increase the field and the current at which this scenario is developed with the intent of reaching 1.6 MA current at full field (2.1 T) in DIII–D in 2003. Further increases in long pulse EC power and the magnet pulse length will culminate in DIII–D being the laboratory for the study of the moderate pulse advanced tokamak called for in the FESAC goals.

1.6.5. NEOCLASSICAL TEARING MODES

While the highest performance AT scenarios in 1999 ran first into the resistive wall mode limit, the longer duration plasmas from 2000 first encountered the NTM limit. The primary strategy of the high bootstrap fraction thrust is to avoid the NTM by retaining a high minimum q from early in the discharge by use of ECCD. However remarkable progess was made in the area of NTMs in the 2000 campaign. Complete stabilization of the 3/2 NTM was achieved with local ECCD, two years ahead of our planned schedule. The work in the NTM area will move more aggressively into stabilization of NTMs with ECCD with applications made to our AT integration thrusts as appropriate.

1.6.6. EDGE STABILITY AND PEDESTAL STRUCTURE

Our AT scenario in 1999 surprisingly went smoothly into an ELMing H–mode edge without encountering the terminations of high performance from edge instabilities prominent in most previous AT efforts on DIII–D. We have developed a detailed understanding of the edge instabilities involving second stable ballooning access afforded by the edge bootstrap current. In 1999, our research thrust on edge instabilities made progess on developing methods to actively intervene in the edge stability situation. However, since our primary scenario was not limited by edge instabities in 1999 and since runtime was very limited, we did not allocate specific runtime to the edge stability thrust in 2000. Another factor in this decision was that we are developing a lithium beam based diagnostic to measure the edge current density and the availability of that diagnostic will illuminate the further study of the edge instabilities in 2001 and 2002. We have decided to allocate specific runtime in 2001 to the edge plasma stability issue with a new focus on the structure and physics of the edge pedestal. The height of the edge pedestal is key to confinement projections to future machines given a stiff transport model for the plasma interior. In 1999, we discovered, and confirmed in 2000, the quiescent H–mode (QH) edge. This plasma edge has the beneficial H–mode pedestal but does not have the bursting edge instabilities (ELMs). There is some continuous edge instability which provides density and impurity control.. This plasma edge is almost ideal and it is mandatory to investigate the physics of this important discovery in 2001 with the emphasis that a thrust affords.

1.6.7. INTERNAL TRANSPORT BARRIER RESEARCH (1999 Progress)

Our research thrust on internal transport barriers is aimed at longer term optimization of AT scenarios. Theory work has pointed to ultimately very advanced states of tokamak performance with nearly 100% bootstrap current in a very hollow profile with a peak near the outer edge of the plasma produced by a broad pressure profile with a transport barrier near the plasma edge. In the long run, it will be desirable to move the transport barrier location to a large radius. Very exciting exploratory work on ITBs was done in 1999 using counter injection to alter the radial electric field profile to affect the E×B turbulence shearing rate with the result of moving the radius of the foot of the transport barrier from $\rho \sim 0.4$ using co-injection to 0.6 using counter injection. Favorable results were also obtained using neon to lower turbulence growth rates. Exploratory results on using inside-launch pellet injection to form transport barriers were also obtained. This work is important to the long term since relatively more bootstrap current can be obtained from a density gradient than from a temperature gradient, within an overall stability constraint on the pressure gradient.

1.6.8. INTERNAL TRANSPORT BARRIER RESEARCH (2000 Progress)

In 2000, in this thrust area, we achieved the combination of the QH–mode plasma edge with the internal transport barrier, resulting in a mode we have dubbed the quiescent double barrier (QDB) mode. This mode showed the potential for long duration plasmas owing to the effective density and impurity control afforded by the QH–mode edge and the fact that the ITB was maintainable stably for several seconds. This mode also showed potential for a high performance candidate, reaching the same β_N*H_{89P} = 7.5 as the high bootstrap fraction scenario. In 2001, we will seek to explore whether the parameter space for the QDB mode is sufficiently broad to enable it to become its own candidate scenario for an AT plasma. We have shown in Fig. 2 the possibility of the QDB scenario evolving into another AT scenario. However this approach needs to define a path for higher bootstrap fraction. That path probably involves higher minimum q and hence this scenario may merge into the high bootstrap fraction scenario in the longer term. This scenario also needs to understand what physics role the counter-injection is playing. If the physics has to do with the edge electric field structure and hence edge orbit loss or rotation, then low energy counter neutral beams may suffice and be technically possible in future machines; if not, then some way of substituting for the effect of the counter beam will be needed. This scenario also has a near term issue of possible high Z concentrations near the axis from neoclassical inward transport owing to the high density gradient from the internal transport barrier. Notwithstanding these challenges, the profiles that have resulted from this QDB operation are so close to what is imagined for the ultimate potential of the tokamak, that this line of research must be pursued.

1.6.9. HIGH INTERNAL INDUCTANCE SCENARIO

Another longer term AT research objective is to open a second major line of AT work, the high ℓ_i thrust, starting in 2003. The high ℓ_i scenario is also a credible path to an Advanced Tokamak future. We intend to restart work on this scenario in 2003 with the initial physics investigations of active stabilization of the sawtooth instability by fast waves followed by the development of scenarios using fast wave, electron cyclotron, and neutral beam heating and current drive power in later years. This next major research line can be expected to develop a logic diagram as complex as that shown in Fig. 2 for the high bootstrap fraction scenario.

1.6.10. FACILITY CAPABILITIES

The Advanced Tokamak facility capabilities needed to accomplish this research are shown in Fig. 3.

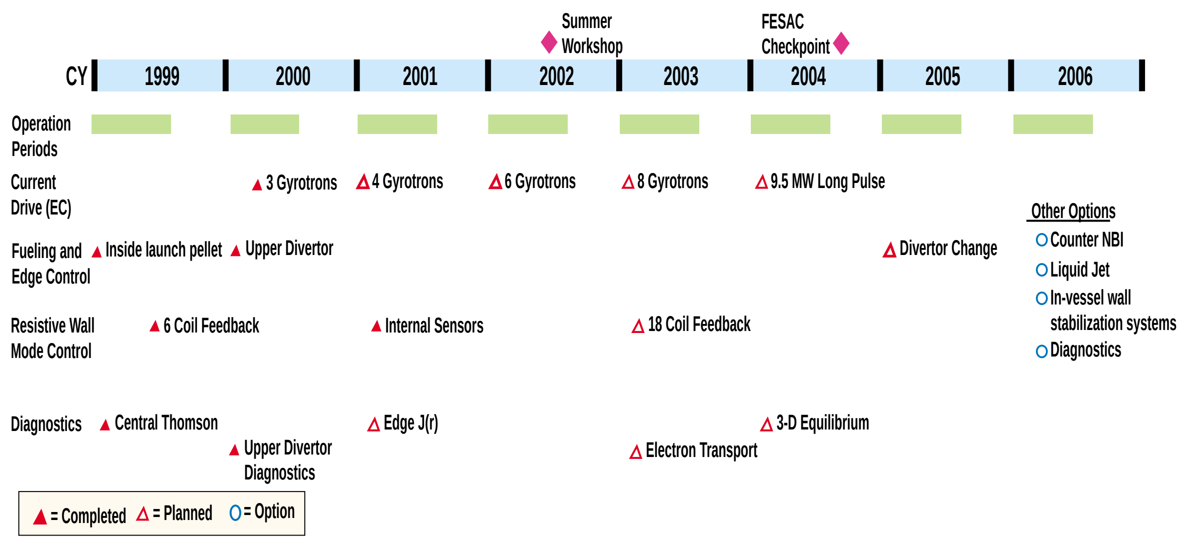

Fig. 3. The DIII–D Advanced Tokamak Program facility capabilities.

Electron Cyclotron Systems

The key hardware capability being implemented is high power, long pulse gyrotrons. The new gyrotrons are nominal 1 MW output power and are equipped with diamond windows for 10 second operation in DIII–D. Three of these new gyrotrons from CPI are coming to DIII-D. The first two production tubes passed factory acceptance tests (500 kW for 10 seconds) and were delivered to GA. The third production gyrotron had reached 600 kW for 5.6 seconds when its output window developed a crack. Installing a new window will delay this tube until after the 2001 experiment campaign. Three diamond window gyrotrons from CPI were brought into service at GA in 2000. One was an older development tube which was operating well until its diamond window developed a leak. The second CPI tube brought into operation was the first of the new CPI production tubes. That tube met its milestone for operation at DIII–D but shortly thereafter also developed a window leak. The window leaks on this tube and the older development tube were owing to corrosion of their aluminum window braze material in the water coolant. The second two production gyrotrons have a gold-copper braze and so should avoid the corrosion problem. Nevertheless, the window crack on the third production gyrotron has galvanized a large effort to better understand these diamond windows. The Virtual Laboratory for Technology is assisting.

Apart from the window difficulties, the basic gyrotrons themselves have performed well. Experiments in the year 2000 were conducted with three gyrotrons. Two Russian gyrotrons with two second pulse length had been obtained from TdeV and brought into service and the second CPI production gyrotron was used also. Experiments in 2001 will be conducted with four gyrotrons. One will be the CPI production unit #2 which is currently running 800 kW for 2 seconds; two will be the Russian gyrotrons from TdeV; the fourth will be an older Russian gyrotron that we simply did not operate last year. This complement of EC sources will enable us to carry out the year 2001 research program. The two additional new CPI production tubes will become available in the year 2002, so we can then begin experiments attempting the high bootstrap fraction scenario at the higher parameters identified as six tubes in Table 3 of Section 2. To enable the AT studies at longer pulses and full field and beyond our intermediate scenarios, two more production unit gyrotrons are planned for 2003. In 2004, the EC system power will be brought to the full power called for in the scenarios described in Section 2.2.3 by the installation of three higher power (1.5 MW) gyrotrons being developed by the Virtual Laboratory for Technology.

An interesting recent suggestion for use of the eventual EC system is as a plasma burn simulator. Alpha particles generated in future burning plasma experiments basically contribute two important effects: possible kinetic instabilities and electron heating. The electron heating effects on profiles and general discharge control can be simulated by the use of electron cyclotron heating. This requires partitioning the EC power into that used to create an Advanced Tokamak scenario and that used to simulate the growing alpha heating of electrons as burn develops.

Particle Control Systems

For density control, the upper divertor private flux baffle and inner leg pump which were installed at the end of 1999 gave us in 2000 the required density control for high triangularity plasmas using the upper pumps or for low triangularity plasmas using the lower pump. Carbon levels in the plasma were also reduced a factor of 2. The UEDGE code had predicted this reduction in carbon owing to the baffling of the divertor. However, we also machined the inner wall and private flux baffle divertor surfaces to a true circle (as opposed to the previous faceted surfaces) and closed tile gaps and precisely aligned their edges to 0.1 mm accuracy. These engineering improvement eliminated hot spots at tile gaps and possibly were the cause of a reduced carbon source. Experiments in 2001 will seek to resolve whether the physics or the engineering lowered the carbon in the plasmas. We do not plan another major divertor change in the near future. As noted above, the plasma shapes compatible with this divertor, while still at triangularity 0.7 are more triangular than any other tokamak, possibly support about a 10%–15% lower normalized beta limit than the previous triangularity 0.9 plasmas. We feel the best strategy to make a major gain in AT performance is to use the effective density control we have with this divertor and push on to the ECCD work and the wall stabilization work. We have shown a possible divertor change for the 2005 campaign after the necessary buildups in the EC and wall stabilization systems are finished. The pellet fueling capability, extended to inside launch in 1999, proved valuable in the pellet fueling capability, extended to inside launch in 1999, proved valuable in triggering internal transport barriers in the density channel. This pellet fueling capability will be further utilized to explore transport barriers and to extend the AT thrust beyond the intermediate scenarios (post 2001).

Wall Stabilization Systems

Stabilization of the resistive wall mode by non-axisymmetric feedback coils is the critical path item that large portion of the tokamak operating space that is predicted to lie above the free boundary beta limit. We have carried out basic wall stabilization research using a set of 6 picture frame coils external to the DIII–D vacuum vessel. These coils were originally installed to study low order non-axisymmetric error field effects and to provide approximately dc correction of those error fields. Faster power supplies were added to enable studies of feedback stabilization of resistive wall modes. We also added a set of 18 picture frame sensors of radial magnetic flux outside the vacuum vessel. These sensors enable various feedback schemes such as the smart shell, mode control, and the fake rotating shell. This research line has shown the existence of the resistive wall mode and the ability of this feedback hardware to affect the growth and stability of the mode. The proof-of-principle research awaits for completion sustainment of a long duration plasma clearly above the no-wall beta limit in 2001. New sensors inside the vacuum vessel have been installed and must be incorporated in the feedback methodologies. These sensors have been predicted to increase the ability of this six coil set to sustain a plasma above the no-wall limit. However, it has long been clear that the optimal mode spectrum for resistive wall mode work requires an expansion of the six coil system to 18 coils with additional coil sets above and below the midplane. For the 2003 campaign, we plan to have expanded the feedback coil set to 18 coils. The more optimal mode spectrum that results is predicted to make possible 80% of the theoretical gain from the no-wall limit to the ideal wall case. DIII–D is unique in the world in this wall stabilization research. It is important that DIII–D do the best job possible with this research line, since DIII–D results will heavily influence what others might subsequently be motivated to attempt. Success in this line will be the key to opening path to AT operation.

Diagnostic Systems

Enhanced diagnostic capabilities will support the evolving AT research plan. A lithium beam diagnostic will be brought on-line during the 2001 campaign to enable measurement of the edge current density profile to significantly enhance the scientific understanding of edge MHD instabilities related to the edge bootstrap current, which we believe opens second regime access. This increased understanding will be important in developing techniques for preventing edge MHD instabilities from terminating high performance AT phases. Although important work remains to clearly identify ITG modes in a plasma, the transport frontier is moving on to the yet smaller wavelength turbulence probably responsible for the residual anomalous electron transport when the longer wavelength turbulence has been suppressed by sheared E×B flows. An initiative in diagnostics for this electron transport is planned. After the complete installation of the resistive wall mode feedback system is completed, a set of diagnostics to enable reconstruction of 3-D, non-axisymmetric equilibria is planned.

Over the three year period 2001–2003, as physics progress pushes out to longer duration AT phases, a set of modest modifications to the thermal capacity of the DIII–D toroidal coil connections and poloidal coil power supplies will be made to bring the pulse length to 10 seconds. The radiative divertor physics will be called upon then to provide sufficient radiative heat dispersal in the divertor to enable 10 second pulses and thereby, to fully integrate AT operation with effective divertor operation.

2. ADVANCED TOKAMAK PROGRAM PHYSICS

2.1. PHYSICS ELEMENTS OF THE PRINCIPAL AT SCENARIOS

The goal of the DIII–D program is to establish the scientific basis for the optimization of the tokamak approach to fusion energy production. This scientific research has many elements, but the principal focus of the DIII–D program toward achieving this optimization is the advanced tokamak program. The advanced tokamak program is aimed at improvement of the tokamak concept towards higher performance and steady-state operation through internal profile modification and control, plasma shape optimization, and MHD stabilization. The dependence of the core performance on the boundary conditions, and the operational regimes envisioned, put more stringent requirements on the divertor and edge plasma, leading to inclusion of divertor optimization and control in any tokamak optimization program.

Two characteristics make the optimization of tokamak performance “advanced”: The inherent one and two dimensional dependence of tokamak performance on the plasma profiles, shape, and boundary; and the requirement to develop solutions that are both multidimensional and self-consistent. The performance capabilities and limitations of the tokamak, and requirements for an energy producing tokamak have long been communicated in terms of global zero-dimensional parameters and largely empirical scaling relations. Chief among these scaling relations are the confinement scaling relations and the scaling of beta with normalized current, known as Troyon scaling. More recently we have discovered, both experimentally and theoretically, that the performance of the tokamak plasma also depends largely on the details of internal plasma profiles, details of the plasma shape, and details of the plasma boundary.

This improvement in our understanding depended critically on the development of new diagnostics to measure the important profile parameters, such as the motional Stark effect diagnostic for measuring the internal magnetic field structure, the charge exchange recombination system to measure toroidal and poloidal plasma flows, and many new turbulence measurements. These new measurements lead to discovery and appreciation of new and important physics phenomena in the tokamak, such as the role of sheared E×B flow, and neoclassical tearing modes.

Equally important to new diagnostic capability is the development of new theories and modeling capabilities to put the transport, stability, and current drive projections on a firmer physics basis. An excellent example of the modeling and theory progress is in gyrokinetic and gyrofluid approaches for physics based transport calculations, and the appreciation of the importance of sheared E×B flow in the predictions.

The self-consistency of the parameters and profiles of high performance plasmas is one of the leading challenges of the advanced tokamak program. As well as the details of both the current density and pressure profile impacting the ideal stability limit and stability to non-ideal modes, at high beta the self-generated bootstrap current is necessarily a major component of the total current. Since the profile of the bootstrap depends on not only the profile of the pressure but of its individual constituents (density, electron temperature, ion temperature, …), the pressure profile and the current density profile are not separable. But, the pressure profile is determined by the transport profiles. In turn, the details of the pressure profile and the current density impact the turbulence growth rates and sheared E×B flow which predominantly determine the transport. In a final advanced tokamak scenario, these interdependencies and complex nonlinear relationships must be fully taken into account and fully integrated. This process greatly benefits from and contributes to the development of a strong fundamental (first principal) physics basis for fusion science.

The DIII–D Advanced Tokamak program aims to develop the best possible operational scenario for fusion energy production using the tokamak. There are many opportunities to make improvements, and many complex interdependencies that allow for a multitude of possible advanced tokamak solutions. In this context it is important to recognize that our rapidly developing understanding and new innovations can lead to scenarios that we do not now envision. So, in developing the “scientific basis for optimization of the tokamak’’ we consider of paramount importance to maintain an attitude of research that is open to new discoveries and continual improvements. We therefore try to plan a DIII–D program that is not only targeted toward testing specific scenarios, but is also optimally positioned to take advantage of new discoveries and innovations. This translates directly into developing diagnostic and control capabilities that are flexible and versatile.

To make significant progress in our research, it is important nevertheless to focus on testing specific scenarios while being alert for discovery. It is important to set aggressive and measurable goals (targets) toward which to focus our efforts. We take our best present understanding of the physics and our best vision of the future embodiment in an energy producing system and develop scenarios, which we can test experimentally in the DIII–D device.

Consideration of physics and energy production lead us naturally to two principal steady-state advanced tokamak scenarios. These two scenarios are negative central magnetic shear (NCS) and high internal inductance (high ℓ_i). These two scenarios do not encompass all the known approaches to tokamak improvement, but rather provide some focus to the challenges that confront us. The profiles and conditions of the two scenarios are quite different, but it is recognized that a fully optimized scenario might lie somewhere in the space between the two.

The viability of a tokamak as an economically and environmentally attractive power plant requires both sufficient energy confinement time, τ_E, for ignition margin, and sufficient volume average toroidal beta, $\beta_T = 2\mu_0 \langle p \rangle / B_T^2$, for adequate fusion power density. Further improvements in the tokamak reactor concept can be made if these improvements in β_T^{max} and τ_E are obtained in steady-state discharge conditions (Kikuchi 1993). We are seeking scenarios that have the potential for high beta, high confinement consistent with steady state, and consistent with divertor scenarios that can provide adequate heat removal, particle and helium ash control, and impurity control.

A minimum necessary condition for an attractive fusion energy producing system is high energy gain. Some insight into possible operational scenarios is obtained by considering the energy gain for a steady-state system, given by Eq. (1):

$$Q \propto \frac{P_{fus}}{P_{CD}} = \frac{P_{fus}}{\frac{n\,R\,I_P}{\gamma_{cur}}\left(1 - f_{BS}\right)} \propto \frac{\gamma_{cur}\,\varepsilon_{eff}\;\beta_N^2}{n\;q\;\left(1 - \xi\sqrt{A}\,q\;\beta_N\right)} \quad . \tag{1}$$

In Eq. (1), P_{FUS} is the fusion power, the P_{CD} is the current drive power, γ_{CUR} is the current drive efficiency, ε_{EFF} is the effective inverse aspect ratio, **A** is the aspect ratio, q is the safety factor at the plasma edge, β_N is the normalized beta and f_{BS} is the fraction of the total current that is the self-driven bootstrap current. In any steady state scenario, care must be taken to minimize the current drive power required. One can view two separate approaches (NCS or high-ℓ_i) to minimizing this current drive power: (1) maximize the bootstrap fraction, or (2) maximize the efficiency of current drive (γ_{CUR}/nI). If the bootstrap fraction becomes a major fraction of the total current, the current profile becomes naturally hollow with the maximum off-axis and the central portion of the plasma has negative central shear, NCS. The bootstrap fraction is further increased by increasing the minimum value of q, q_{min}, and moving the radius of the q_{min} to larger radius. If the emphasis is placed on increasing the current drive efficiency, it is natural to drive the current on axis where the temperature is highest (current drive efficiency is proportional to electron temperature) and where the effects of trapping are minimal. Axial current drive leads naturally to peaked current densities, with large positive magnetic shear in the outer plasma region. A schematic of the resultant current profiles is shown in Fig. 4. The actual current profile for these two cases depends on establishing consistency among the profiles, stability, and transport.

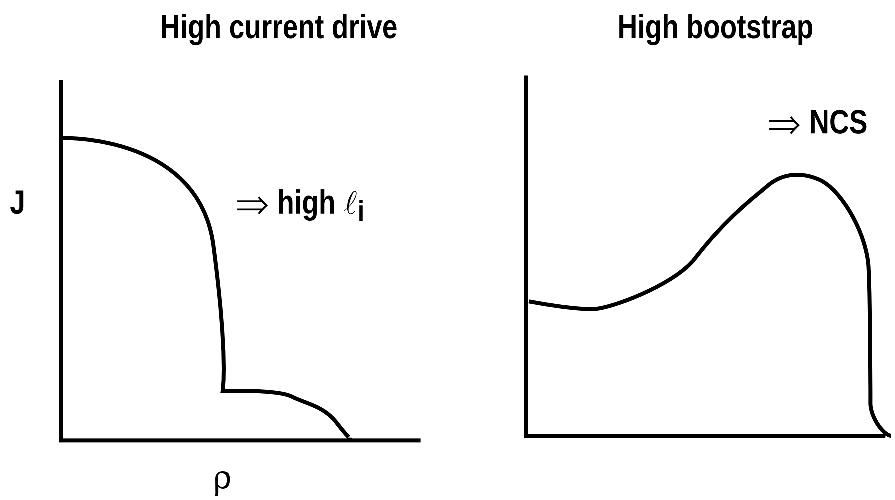

Fig. 4. Steady state considerations also lead to two "natural" current profiles.

Unless the NCS scenario has fully 100% bootstrap driven current, it is important to maintain relative high current drive efficiency in both the NCS and high ℓ_i scenarios.

The need for high current drive efficiency pushes steady state operational regimes to higher temperature and lower density than might otherwise be the optimal in a Ohmically pulsed scenario. The higher temperature and lower density, impose new challenges for heat removal and impurity control for the divertor. This lower density, higher temperature operation motivates the inclusion of divertor optimization as an important element in the DIII–D AT program.

Simple physics considerations also lead to the same operational scenarios, (1) NCS and (2) high ℓ_i. We show in Fig. 5, the general dependence of ideal ballooning stability and ion temperature gradient (ITG) driven instabilities on the magnetic shear, $S_M = \rho/q\ (\partial q/\partial\rho)$. These general dependencies, known for a long time, clearly show that both low or negative magnetic shear and high magnetic shear are favorable for stability of ballooning modes and ITG modes. These physics considerations lead to the same two general classes of scenarios given above; (1) low or negative shear → NCS, and (2) high positive shear → high ℓ_i. It is worth noting that the magnetic shear (in the large aspect ratio circular limit) observed experimentally in Ohmically driven discharges is near 1, nearly the most unfavorable value for ballooning and ITG mode stability. So one might expect that the ability to modify the current profile toward either larger positive or negative magnetic shear would lead to positive benefits.

2.1.1. GENERAL NCS CONSIDERATIONS

The NCS scenario has the potential for a high bootstrap fraction at moderate q: the bootstrap fraction can approach unity at q_{95} = 5–6. Furthermore, there is the potential for the bootstrap current to be well aligned with the total current, resulting in low, total current drive requirements. The hollow current profile, and the resultant region of negative central magnetic shear derive naturally from the bootstrap current. The bootstrap current is proportional to the square-root of the local aspect ratio times the pressure gradient, both of which go to zero on axis, so that the bootstrap current profile is naturally hollow. In addition, the higher axial q and lower poloidal field in the core have the effect of increasing the total bootstrap fraction. The high bootstrap fraction results in lower total current drive, but highly localized, off-axis, precision current drive is needed. Electron cyclotron current drive is well suited for the precise off-axis current drive needed. Because of the potential of the NCS scenario with respect to fusion energy, we have chosen it as the leading scenario on which to focus.

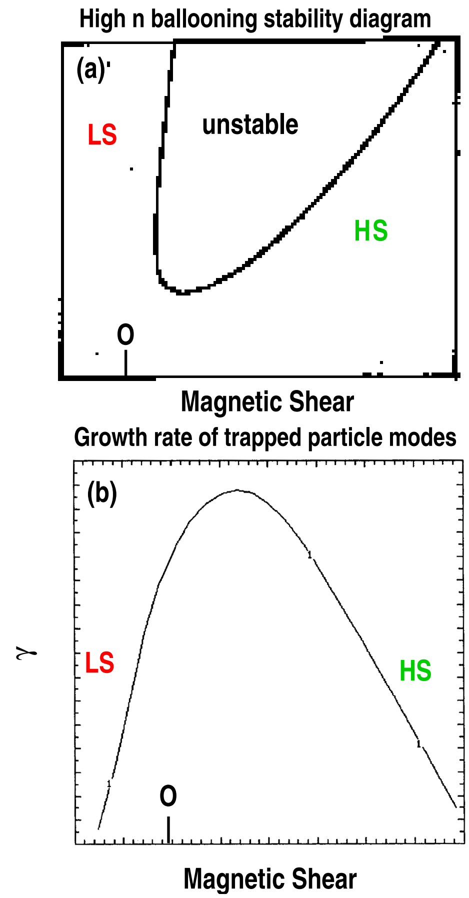

Fig. 5. Both low magnetic shear (LS) and high shear (HS) are favorable for: (a) higher beta, (b) reduced turbulence and reduced transport. Magnetic shear is $s \propto R/B_T^2 q^2\, d\rho/dr$.

The NCS scenario does have some very specific challenges. The first challenge is stability. Stability to ballooning modes is a necessary condition for achieving high beta, and therefore an important consideration. The NCS scenario avoids ballooning mode limitations because the region of high pressure gradient is in the region of low or negative shear, where there is access to the second regime and no limiting pressure gradient is calculated. Furthermore, the negative shear region is stabilizing to neoclassical tearing modes where the pressure gradient is expected to be large, and if the minimum value of q is above 2, the absence of low order rational surfaces should further diminish the importance of the modes.

There are several MHD instabilities that remain a challenge to the NCS scenario. Strong pressure gradients in the region of negative shear can be destabilizing to resistive interchange modes. Modeling indicates that these modes are stable if the magnitude of the negative shear is kept modest. Double tearing modes are calculated to be unstable as a consequence of the double value of q. However, these are rarely observed in the experiment, and modeling indicates modest rotational difference in the plasma between the two surfaces (as observed in the experiment) is sufficient for stabilization.

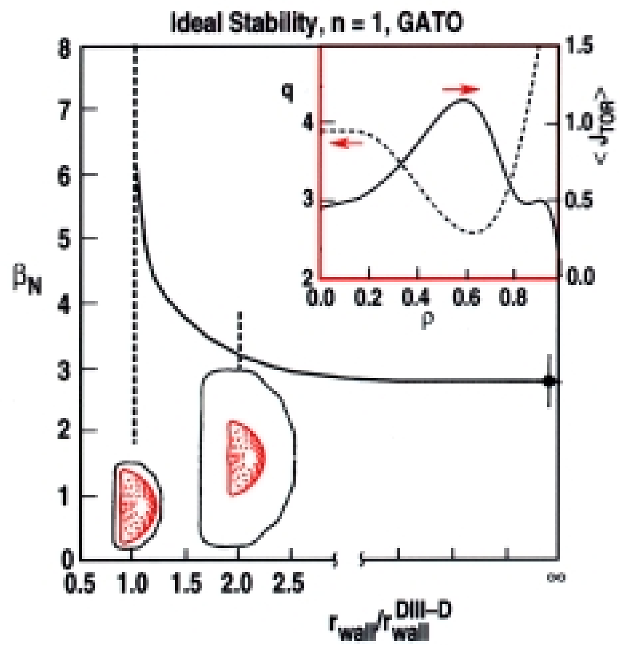

Fig. 6. Maximum stable beta increases for closer wall position: ideal n=1 stability using DIII–D plsma shape and DIII–D wall. Insets are typical current density and q profile (Taylor1995).

The NCS scenarios have quite low ℓ_i and are generally unstable to the external/global kink, in the absence of a conducting wall. However, the broad current density profile, broad pressure profile, and strongly shaped plasmas couple very strongly to a nearby wall. Modeling indicates that $\beta_N > 5$ stable to n=1 and 2, is easily obtained if a conducting wall is located at $r_W/a <$ ~1.5. The modeling calculations for n=1 are shown in Fig. 6. However, the real wall is resistive and the plasma is subject to the resistive wall mode. The stabilization of the resistive wall mode then is key part of validating and optimizing the NCS scenario. The DIII–D program is taking two approaches to stabilization of the resistive wall mode; passive stabilization with a rotating plasma in the presence of a resistive wall, and active feedback stabilization with non-axisymmetric external coils.

It is important to note that reasonably high beta values can be calculated for the NCS scenario without a conducting wall; β_N values <4 are calculated, very similar to the high ℓ_i scenario. So if wall stabilization proves not to be so attractive in a reactor embodiment, there remain attractive NCS and high ℓ_i scenarios.

For the NCS scenario, perhaps the most challenging physics lies in the consistency of the profiles. A range of current density and pressure profiles can be identified that are consistent with high beta stability. In particular, it can be shown that broad pressure profiles are required for high beta stability and alignment of the bootstrap current (Fig. 7). However, the combination of the q profiles, pressure profiles, and rotation (E×B) profiles often result in transport reduction and often the formation of a clear internal transport barrier that leads to pressure peaking that are not compatible with high beta.

Discharges in DIII–D can have an internal transport barrier, ITB, with no edge transport barrier (L–mode NCS), a strong edge transport barrier, (H–mode NCS), and a self-regulating edge barrier with an internal transport barrier (ELMing H–mode NCS). (However see Section 2.4.2 for a discussion of the recently discovered quiescent double barrier mode.) These three cases have different challenges with respect to high beta and self-consistent solutions. The L–mode NCS has a very weak pressure gradient in the outer portion of the plasma. The key challenge for L–mode NCS is to move the transport barrier to larger major radius to achieve higher beta [as shown in Fig. 7(a)] and to obtain good bootstrap alignment. It is also important that for stability, the width of the transport barrier region not become too narrow as indicated in Fig. 7(b). In general, peaked pressure profiles that result from an ITB at small radius result in a low stability limit as shown in Fig. 8. The ELMing H–mode NCS and the H–mode NCS both lead to broader pressure profiles and the potential for high beta with an ITB. For the ELMing case, the repetitive ELMs provide a seed for neoclassical tearing modes, and the higher pressure gradient in the positive shear region make the neoclassical tearing modes unstable. For H–mode NCS, a clear strong barrier exists near the boundary, and the plasma is subject to low n kinks associated with the high edge pressure gradient and high edge current density. High beta, broad profiles, and strong shaping cause strong harmonic coupling and these edge driven modes are no longer localized to the edge. The key challenge for the H–mode NCS scenario is to understand how to moderate the edge and avoid the edge instability or its strong coupling to the core.

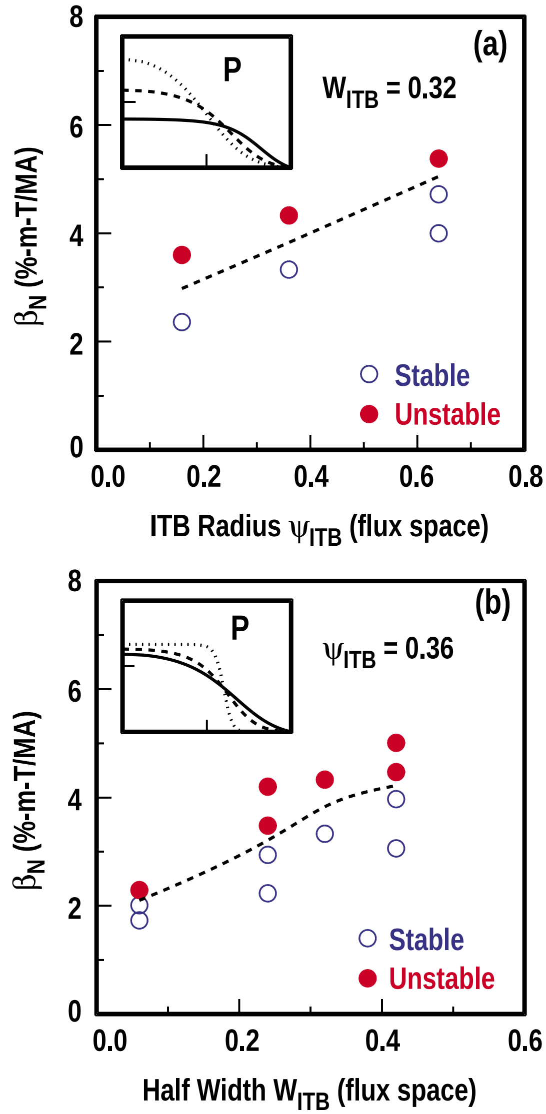

Fig. 7. Stability limit improves with internal transport barrier width and radius. Fixed shape DND, q_{95} = 5.1, q_0 = 3.2, q_{min} = 2.2; hyperbolic tangent pressure representation; ideal n = 1, wall at 1.5a.

A physics understanding of the edge instability is unfolding and we are developing techniques to modify and control these instabilities. In DIII–D the edge pressure gradient is limited by moderate n (2 ≾ n ≾ 9) kink/ballooning instabilities. The magnitude of the resulting instability (the edge localized mode) increases with decreasing toroidal mode number of the instability. These modes are driven by both the locally high pressure gradient and the locally high edge current density (the edge current density is the bootstrap current driven by the edge pressure gradient). The edge region is typically in the region of second stable access to infinite n ballooning modes, and local pressure gradient is significantly higher than that expected from the first regime limit. High squareness shaping eliminates the access to second regime stability and leads to much smaller and more frequent ELMs (presumably at higher n) more compatible with an internal transport barrier. Edge impurity radiation reduces both the edge pressure gradient and the edge current density and also gives smaller and more frequent ELMs. Both plasma shaping and edge impurity radiation are being evaluated as techniques to control the edge instabilities, and better measurement of the edge current density with Li beam polarimetry is being pursued in order to better quantify the edge stability models.

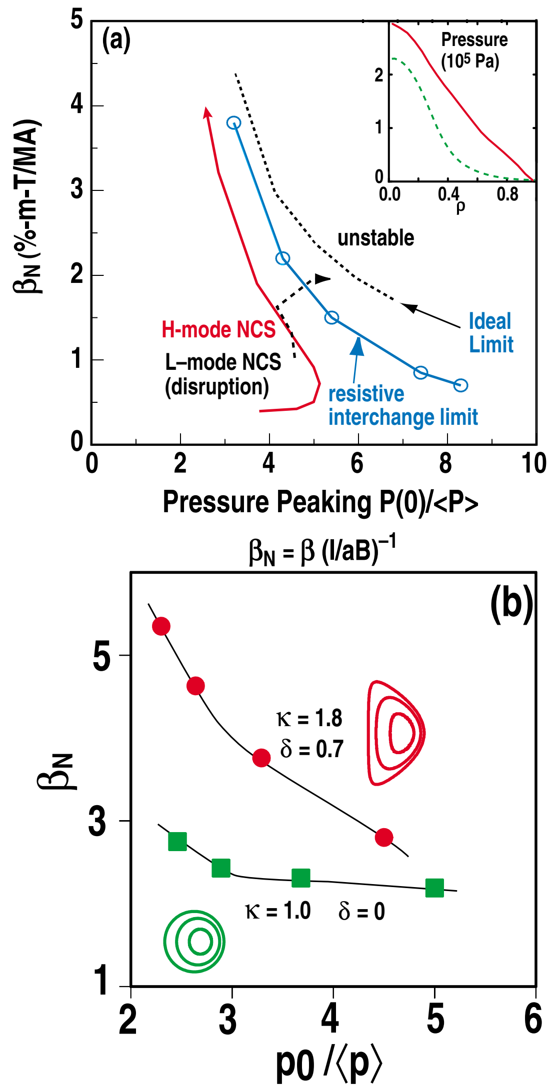

Fig. 8. Higher β is obtained with broad pressure profile (a) normalized beta vs. pressure peaking. Ideal and resistive limits are from generated equilibria similar to the experimental. Dashed trajectory is for an L–mode NCS discharge, solid trajectory is for H–mode NCS discharge (Lao 1996). Insets are exp. pressure profile just prior to disruption.(b) β_N vs. pressure peaking for D shaped and circular shaped equilibria, q_0 = 3.9, q_{min} = 2.1, q_{95} = 5.1, rw/a = 1.5 (Turnbull 1996).

A sound physics understanding of the reduced transport in the NCS discharges, and of the transport barrier formation, is developing based on sheared E×B flow stabilization of microturbulence. An extremely rich variety of physics effects provide for exciting and interesting fusion science research, as well as opportunities for control of the transport and transport barrier. The ability to vary the location of the internal transport barrier and to control the magnitude of the local pressure gradient can allow us to generate pressure profiles consistent with high beta stability and bootstrap alignment; see Fig. 7.

2.1.2. GENERAL HIGH-ℓ_i CONSIDERATIONS

A significant experimental basis for a high ℓ_i high performance scenario exists. A number of tokamaks have observed experimentally that the maximum achievable beta increases with internal inductance, and the DIII–D experimental program has established the scaling relation $\beta_{max} = 4\ \ell_i * I/aB$.(Taylor 90 IAEA). This relation has been supported by a large number of experimental results from other tokamaks. It has also been shown to be consistent with theory and modeling results (Lao 1991), at least for a class of equilibria generally consistent with Ohmically driven current profiles. It has also been shown in a wide range of experiments that the energy confinement time increases with ℓ_i. The increase in confinement has been shown to be a consequence of an increase in magnetic shear and a consequence of an increase in the E×B flow shear. There exists a positive feedback mechanism between the two effects. These high confinement and high beta results have to date been achieved transiently by ramping down the plasma current or by expanding the plasma size.

Self-consistency of the profiles in steady state does place limitations on the high ℓ_i scenario. Maintaining q_0 slightly above unity and avoiding the m/n = 1/1 sawtooth instability has been observed to be a necessary condition in achieving high performance in many tokamaks. We will make the most peaked current density profile possible (highest ℓ_i) consistent with ballooning stability and resulting allowable pressure profiles in the following way. The current profile will consist of the driven seed current and the bootstrap current. The driven seed current will have a top-hat form and is located in the core, with the limitation that $q_0 > 1$. The total current density is equal to the maximum of the local current density and the local bootstrap current. The pressure gradient is limited to remain below the ballooning limit. The resultant current density profile is shown in Fig. 9, and the internal inductance is limited to $\ell_i \sim 1.1$. The maximum beta stable to ideal ballooning modes in such a case is $\beta_N < 4$ (for $\kappa = 1.8$, $\delta = 0.7$ equilibrium), and the maximum bootstrap fraction is limited to approximately 60% at q ~ 7. This scenario is an attractive advanced tokamak scenario, and we think the physics challenges are not very

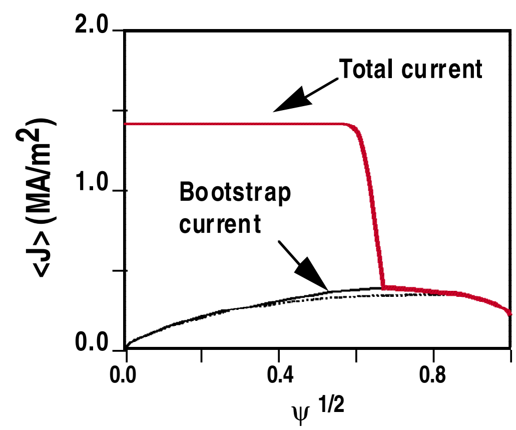

Fig. 9. Self-consistent current profile from high β, high ℓ_i equilibrium $\beta_N = 4$, $\ell_i = 1.2$, $q_{95} = 8$, $q_0 = 1.05$.

demanding. However, because of its bootstrap current limitations and implications on achievable steady state Q, the high ℓ_i scenario is not our leading scenario.

2.2. THE NEGATIVE CENTRAL SHEAR (NCS) SCENARIO IN DIII–D

The principal approach to the AT in DIII–D is the negative central shear regime. This regime has the best set of characteristics to take forward to a steady-state fusion reactor. The hollow current profile is compatible with the high confinement arising from a transport barrier since the off-axis bootstrap current produced by the transport barrier will produce most of the required off-axis current peak. The rest of the non-inductive current can be either on-axis for central q control or off-axis to supplement and align the bootstrap current peak with the required total current profile. The negative central shear q profile and the broad pressure profile resulting from a transport barrier and q_{min} being at large radius are compatible with high normalized beta. Wall stabilization is also needed owing to the closer proximity of the current peak to the plasma edge. This scenario can be made with either the L–mode or H–mode edge. Which is best for stability and confinement is an active subject of ongoing research.

There is considerable flexibility in this scenario in regard to how the plasma edge is managed and how the interior current and pressure profiles are controlled. It is not clear, for example, how much magnetic shear reversal is needed, even to the limit of zero shear.

On three or four separate occasions in the last four years, different people from different viewpoints have constructed AT NCS scenarios for DIII–D using the ONETWO transport code, the stability codes GATO and BALOO, and the transport code CORSICA. We will summarize those scenarios below in order of increasing complexity of the transport modeling rather than in the chronological order in which they were done. They exhibit some different approaches and interests which provide pathways into the variations in experimental approach being currently pursued and we will comment on those pathways into the ongoing experimental program. Recent work on defining optimum scenarios for ARIES-AT also point to exciting long-term directions for the DIII–D research. Recent experimental work from the internal transport barrier thrust is moving in the direction of these ARIES-AT visions. The quiescent double barrier mode (QDB) discovered in 2000 has an edge H–mode transport barrier without ELMs and yet with density and impurity control and an internal transport barrier. The future task is to move the internal transport barrier out toward the edge barrier, possibly merging the two barriers, to realize the ARIES-AT type of profiles.

2.2.1. SCENARIOS USING FIXED PROFILES

The purpose of this modeling exercise was to demonstrate the potential for intermediate advanced tokamak operation goals at intermediate values of plasma current and toroidal field with the view of a phased installation of a ten gyrotron system (nominal 1 MW/gyrotron source with 70% delivered to the plasma) for ultimate operation at full field and current in DIII–D. For this purpose scenarios with 3, 6, and 10 gyrotrons were developed from B_T = 1.6 T to full B_T = 2 T and I_p ranging from 1.0 MA to 1.6 MA.

The starting point in each case was a stable MHD equilibrium with boundary consistent with the full RDP installation. Only the total pressure is important for the equilibrium, but the non-inductive current density calculations require the pressure to beseparated into electron and ion density and temperature. This division is shown in Fig. 10 for the three gyrotron scenario with β_N = 4.

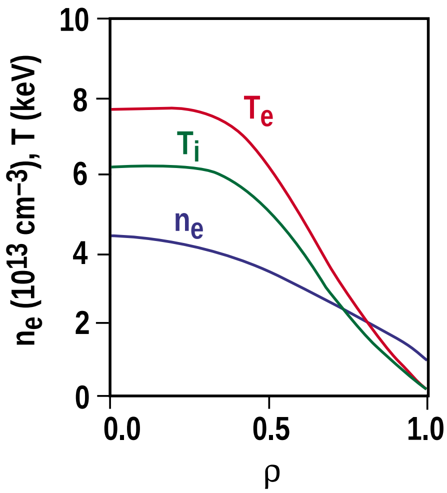

Fig. 10. NCS profiles of temperature and density.

The density profile was chosen to be consistent with pumped ELMing H–mode discharges at higher q_{95}. These are more peaked than the canonical H–mode density profiles normally shown. Very little effort was directed to make an H–mode edge pedestal or consistency between the edge bootstrap current density and the total current density from the equilibrium because the purpose was to show whether the off-axis ECCD was sufficient in these conditions. The level of the line-averaged density was limited by the empirical rule-of-thumb on DIII–D that the ELMing H–mode density can be varied from n (10^{19} m^{-3}) = 3 I (MA) to 6 I (MA). The actual density was maximized consistent with full non-inductive current. The impurity density profile was chosen arbitrarily to give a constant Z_{eff} = 1.5 across the plasma.

The temperature profiles were chosen to be a constant fixed ratio across the entire plasma. The transport code was run in analysis mode to derive both the local transport coefficients and the global confinement relative to the ITER-89P scaling law. The local transport coefficients were checked to ensure the ion diffusivity was at or above neoclassical and near the electron diffusivity.

The same source calculations in the transport code also provide the non-inductive current densities due to NBI, bootstrap, and ECCD. The scenario was iterated to give zero net ohmic current, not zero ohmic current density at all radii. Again, the goal of our modeling at that time can be seen directly by examining Fig. 11 which shows the total current density from the equilibrium and the non-inductive current densities calculated using the profiles in Fig. 10. It is clear that 2.3 MW of EC power delivered to the plasma under these conditions supplies sufficient current at the half radius to maintain the off-axis current density in conjunction with the bootstrap current. A resistive evolution could have been done and would have resulted in a less reversed q profile, but the degree of negative central shear is not believed to be an essential feature of this scenario.

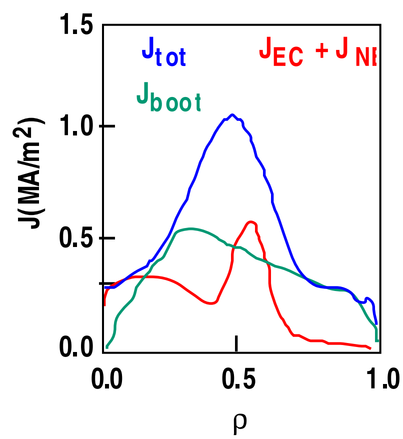

Fig. 11. Contributions to a hollow current profile.

Fixing the profiles is obviously equivalent to fixing the target β_N and H factor for the scenario. These calculations do represent first principles evaluations of where to place the RFCD and the efficiency of the RFCD. These calculations are of value in determining the rf and NBI power levels needed to make the target scenario in terms of current drive and assuming the target values of β_N and H. Scenarios at increasing plasma current and field were developed.

Two scenarios are summarized in Table 3. We have labeled these scenarios by the number of gyrotron tubes we believe we will need to carry them out. These scenario gave focus to the effort in Thrust Area 2 in 1999 to develop transiently the plasma described in the four tubes column. Good progress was made in 1999 on this scenario as shown in Fig. 12. This discharge is an ELMing H–mode as shown by the D_α trace, Fig. 12(f). The discharge is quasi-stationary for ~2 s or 16 τ_E and has a $\beta_N H$ product of 9. β_N is just below 4 and is larger than the nominal no-wall limit of 4 x ℓ_i, Fig. 12(c). Small recurring resistive wall modes were observed and are the cause of the periodic drops in β_N. The plasma described in the 6 tubes column is our principal near-term objective and certainly requires an effective wall stabilization feedback system.

In order to obtain sufficient current drive efficiency, these scenarios use low densities, a low fraction of the Greenwald limit and lower than the density achieved in the discharge in Fig. 12. These low densities are below where detached divertor plasmas are found, setting the challenge to either raise the scenario density or to the divertor program to develop ways of making radiative divertors compatible with these AT core plasmas. Density control at least is required from the divertor program to meet these scenarios.

Table 3
Parameters of NCS Scenarios Using Fixed Profiles.

	4 Tubes	6 Tubes
P_{EC} (MW)	2.3	4.5
P_{FW} (MW)	3.6	3.6
P_{NBI} (MW)	4.1	3.8
I_p (MA)	1.0	1.3
I_{Boot} (MA)	0.65	0.9
I_{ECCD} (MA)	0.15	0.2
B_T (T)	1.6	1.75
β_T (%)	4.0	6.3
β_N	4.0	5.3
H_{89P}	2.8	3.5
n (10^{20} m^{-3})	0.32	0.5
n/n_G	0.3	0.4
$T_i(0)$ (keV)	6	8
$T_e(0)$ (keV)	8	9

2.2.2 NCS SCENARIO SIMULATIONS USING DIFFUSION COEFFICIENTS DERIVED FROM DISCHARGES

MHD stability studies of discharges with an internal transport barrier (ITB) show that the stability limit improves with increasing width and radius of the ITB based on a systematic scan of simulated equilibria with model q and pressure profiles. The scenario modeling described in the preceding section began with a total pressure profile consistent with an MHD stable equilibrium and rather arbitrarily divided the total pressure into electron and ion pressure which were then portioned to density and temperature. The scenario modeling described in this section is based on transport coefficients determined from an existing ITB discharge with an L–mode edge which are then scaled to different parameter regimes. To date the studies have been focused on using this approach to achieve the discharge conditions chosen by the method of the previous section.

Time-dependent transport simulations were performed using the ONETWO and CORSICA transport codes. First, measured profiles from an ITB discharge from our 1999 campaign, very similar to that shown in Fig. 12, with B_T = 1.6 T, I_p = 1.2 MA, q_{95} = 5, β_N = 3.7 and H_{89P} = 2.9 were used to calculate thermal diffusivities $\chi_e(\rho)$ and $\chi_i(\rho)$. These calculated diffusivities, with the addition of the ion neoclassical diffusivity shown in Fig. 13, were the baseline model diffusivities used in the time-dependent ONETWO simulations. The target parameters for the simulations are exactly those of the ITB

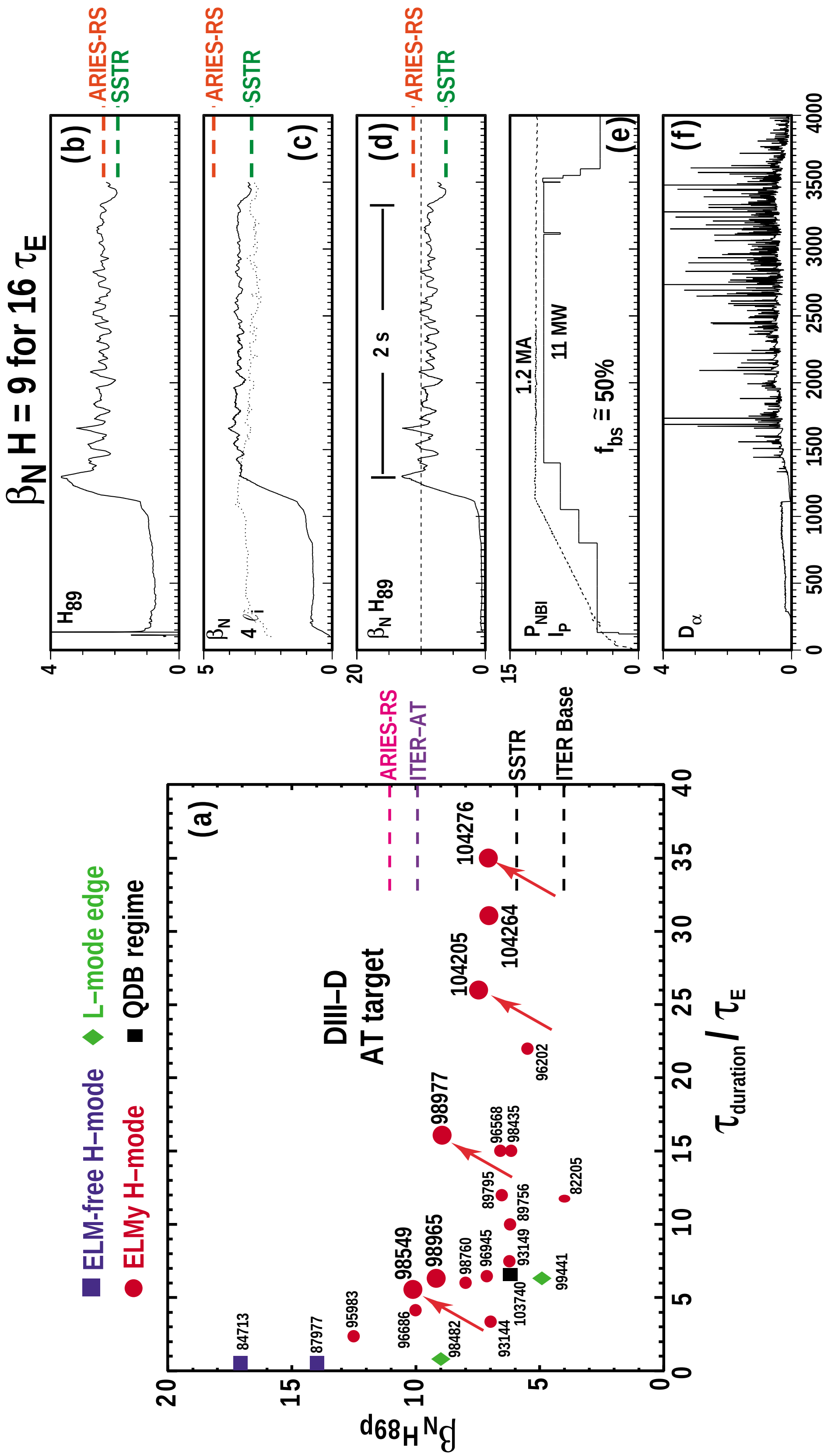

Fig. 12. (a) Recent progress in AT scenario development. The large circles represent 1999 and 2000 progress. (b) Confinement enhancement, H_{89P}; (c) normalized beta, β_N (solid) and 4 x internal inductance, ℓ_i; (d) product β_N, H_{89P}; (e) beam power (solid), plasma current (dashed); (f) divertor D_α.

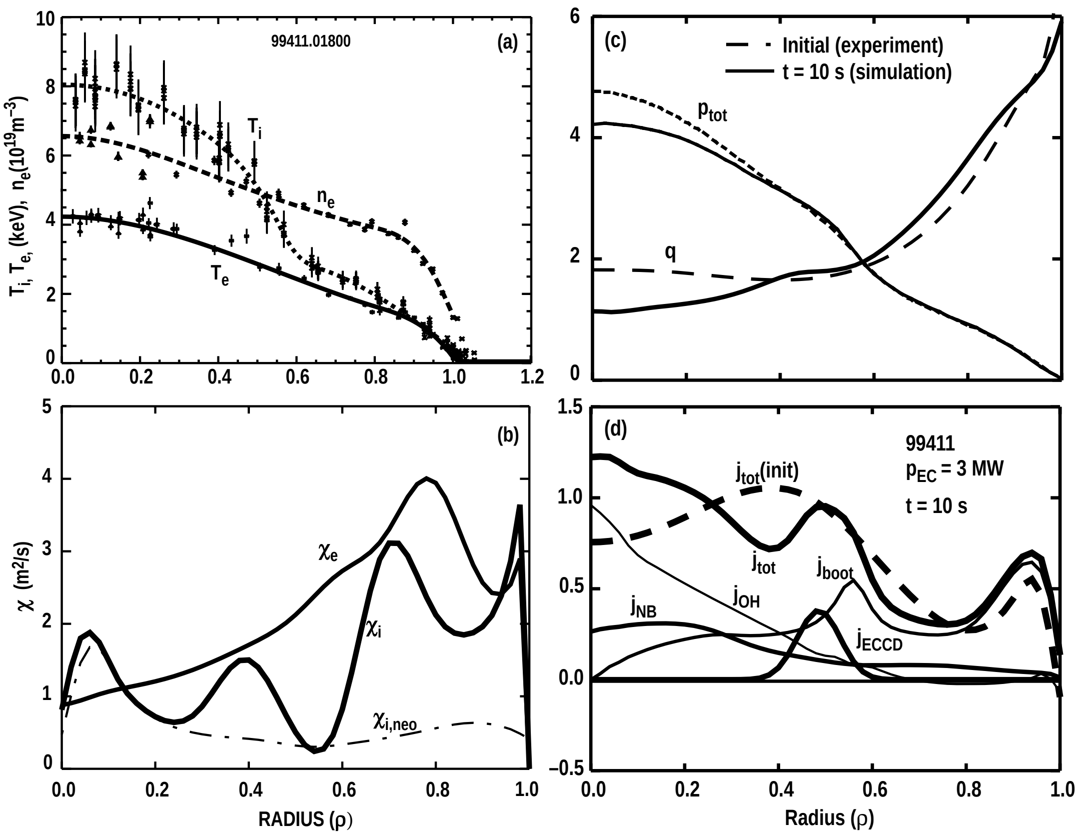

Fig. 13. NCS scenario using empirical transport coefficients. (a) Measured ion temperature, electron temperature, electron density; (b) experimental thermal diffusivities; (c) q profile, total pressure; (d) current profiles.

discharge. CORSICA simulations have concentrated mostly on the sensitivity of results to other transport models.

In the process of performing the transport simulations, a number of iterations are carried out in order to optimize various choices. The ECH launching direction is optimized to align the electron cyclotron current drive (ECCD) profile with the off-axis bootstrap current profile, and to maximize the ECCD efficiency to overcome the dissipating ohmic current profile. We then evolve T_e, T_i, and current density with a fixed density profile for a period of 10 s. The transport profiles (χ_e, χ_i) remain fixed to those from the experiment. We iterate this transport simulation cycle three times, each with the starting profiles taken from those at the end of the previous cycle. In the last cycle, evolution of the MHD equilibrium is also performed. At the end of each cycle, we test the MHD stability with high-n (BALOO code) and low-n (GATO code) stability.

Figure 13 shows the summary of simulations for q(ρ), T_e(ρ), T_i(ρ), and individual components of the current density profile together with a tabulation of the parameters (Table 4), which can be compared with the case from the previous section. The detailed

Table 4
Parameters of L–Mode Edge NCS Scenarios Using Transport Simulations

	4 Tubes
P_{EC} (MW)	0→3.0
P_{FW} (MW)	0
P_{NBI} (MW)	9.2→6.2
I_P (MA)	1.2
I_{Boot} (MA)	0.67
I_{ECCD} (MA)	0.9
I_{OH}(MA)	0.24
B_T (T)	1.6
β_T (%)	4.3
β_N (%)	3.7→3.5
H_{89P}	2.9→2.6
n (10^{20} m^{-3})	0.48
n/n_G	0.48
$T_i(0)$ (keV)	8.4→6.9
$T_e(0)$ (keV)	4.2→4.5
q_{95}	5.0
J_{BS}	55%

case shown in Fig. 13, illustrates that 3 MW of off-axis ECCD at $\rho = 0.5$ maintains $q_0 > 1$ and weak shear for 10 s and, we expect, sustains the high performance phase, allowing more detailed evaluation of AT physics.

With the addition of density control with the two upper cryopumps, we expect significantly more electron cyclotron current and expect a quasi-stationary current profile. To quantify these expectations, we extended the transport simulations described above over a range of densities. In Fig. 14 we show the calculated electron cyclotron current drive versus plasma density. The 1999 discharges were at the right hand border of this diagram. Our desired operating target is shown and requires a reduction of operating density from about 5×10^{19} m^{-3} to 3.5×10^{19} m^{-3}.

In the year 2000 campaign, we had available the completed upper divertor to support pumping of high triangularity (0.7) plasmas. Effective density control was obtained. We were able to produce discharges with a 6.3 second duration (35 energy confinement times) with a β_N*H_{89P} product of 7.5 (Fig. 15). These are the highest performance discharges for this kind of duration yet achieved anywhere. As can be seen in the figure,

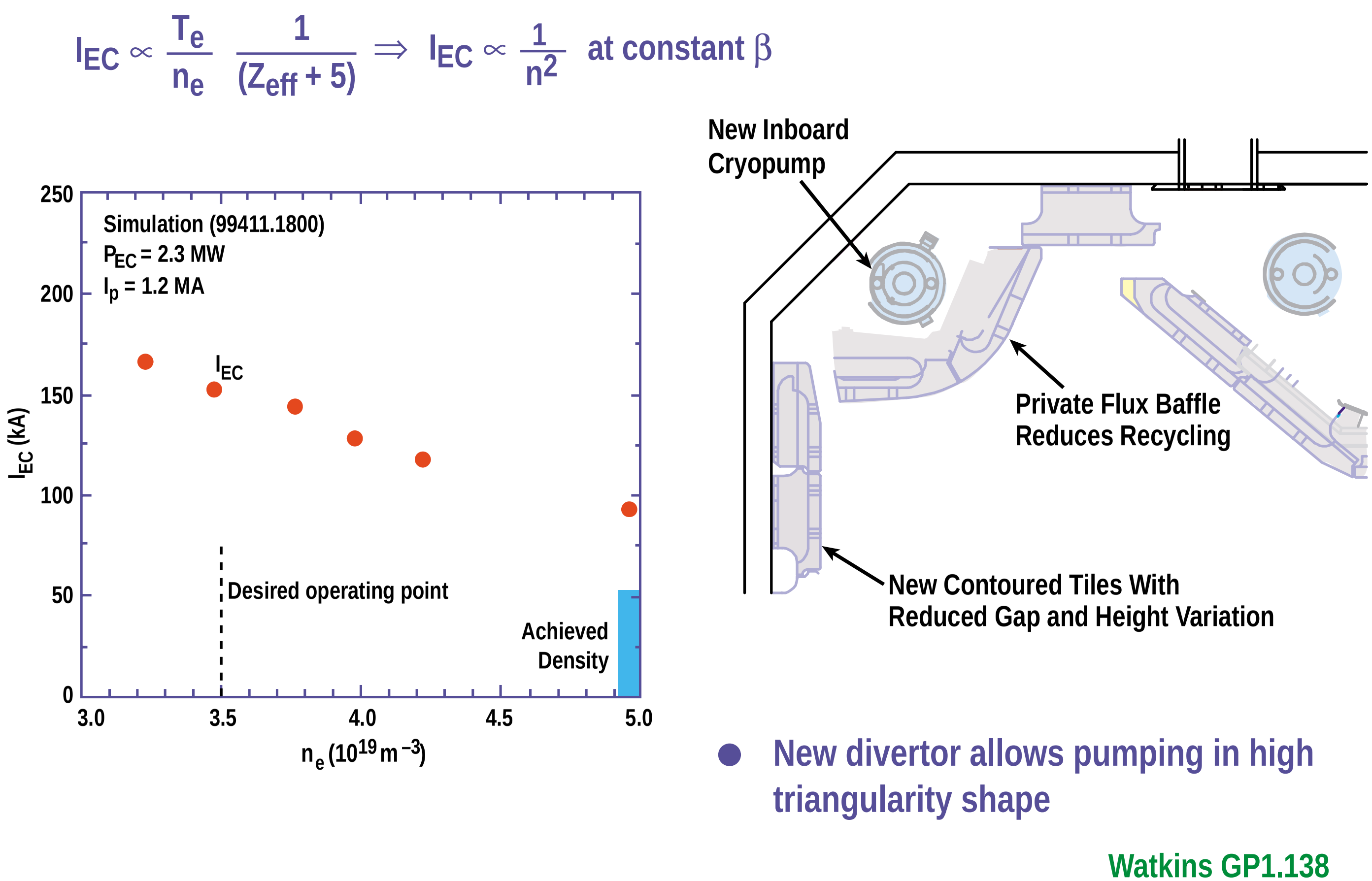

Fig. 14. Density control will maximize the effectiveness of off-axis ECCD.

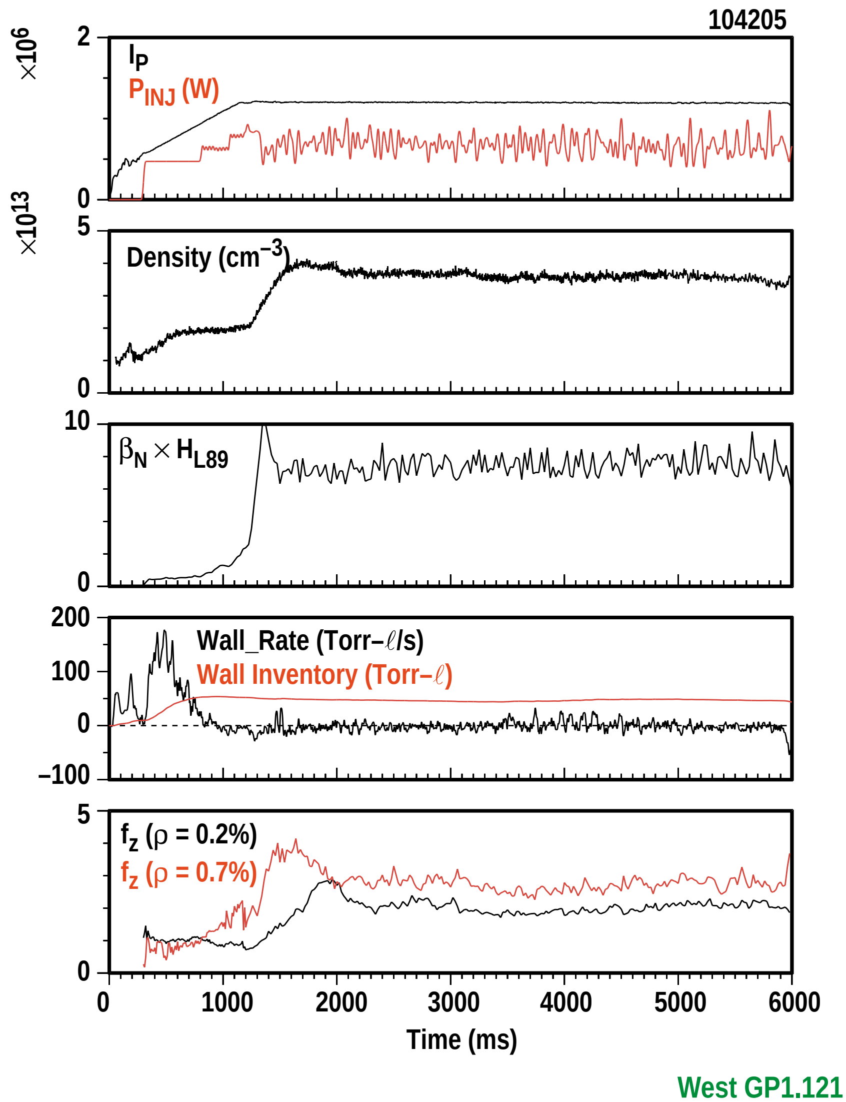

Fig. 15. Density and impurity control has been demonstrated in long-pulse ELMing H–mode discharges with $\beta_N H_{89P} \sim 7.5$ for over 35 τ_E.

the density is controlled at about 3.5×10^{19} m^{-3}, our desired target. The divertor pumping supplied the density control. Wall pumping was negligible and the wall inventory remained constant. Impurity levels were also steady at values about half of what was achievable in the previous year's discharges. The normalized plasma beta was also controlled rock steady at 2.8, a value about 5%–10% below the 2/1 NTM limit, by modulating the neutral beam power. These discharges showed the kind of long pulse integrated performance we are seeking to demonstrate, but we desire eventually higher β_N, H factor, and bootstrap fraction.

However, the 1999 discharge without density control (Fig. 12) used a higher triangularity ($\delta \sim 0.9$) shape than these pumped year 2000 discharges. Achievable values of β_N

for longish durations in 2000 were 10%–15% smaller with the shape optimized for pumping than values achieved in the plasma shapes of 1999. We are still working on whether this apparent shape dependence of the beta limit can be obtained from theory calculations. The lower β_N in these plasmas also meant lower bootstrap fraction f_{BS} ~ 40%.

While it has always been recognized the wall stabilization is the key to opening that large portion of the tokamak operating space that lies above the free boundary beta limit, it is clear for this AT line that wall stabilization is on the critical path to higher β_N. New internal sensors have been installed and must be incorporated in the feedback methodologies. The RWM thrust in 2001 will seek to demonstrate a long duration sustainment of a plasma clearly above the no-wall limit. There may be late in the campaign an attempt to apply the RWM feedback to the benefit of the high bootstrap fraction scenario, but realistically this work is probably best left for the 2002 campaign. For the 2003 campaign, we plan to have expanded the feedback coil set to 18 coils from the present 6 coils. The more optimal mode spectrum that results is predicted to make possible 80% of the theoretical gain from the no-wall limit to the ideal wall case. Success in this line will be the key to opening path to AT operation.

With the density control needed for ECCD established, the next order of business in this research line is to apply ECCD power to counteract the resistive diffusion of the current. Gradual growth in EC power and pulse length in the period 2001–2003 will enable longer pulse sustainment of the AT scenarios. As more ECCD power becomes available throughout 2001–2003, we will increase the field and the current at which this scenario is developed with the intent of reaching 1.6 MA current at full field (2.1 T) in DIII–D in 2003. Further increases in long pulse EC power and the magnet pulse length will culminate in DIII–D being the laboratory for the study of the moderate pulse advanced tokamak called for in the FESAC goals.

2.2.3 NCS SCENARIOS USING MODELS OF TRANSPORT BARRIER FORMATION

Scenarios using models of transport barrier formation have been produced twice. A table of numbers for these scenarios is given in Table 5. These scenarios were constructed by 1-D simulations using the ONETWO code. All of the scenarios lie in the range β_T 5%–11% at full field in DIII–D. They are at plasma currents of 1.6–2.2 MA and employ strong shaping (κ = 2.1, δ = 0.8). They employ a total of 15–20 MW of total heating and/or current drive power, made up of roughly equal contributions of NBI, ECH, and FW. They all employ fast wave heating to achieve a high core electron temperature.Except for Case 2, all these cases require at least 6 MW EC power delivered to the plasma. These considerations lead us to believe that a 10 gyrotron EC system will ultimately be needed to form the NCS plasmas at full field and current in DIII–D.

Table 5
Parameters of Longer Range DIII–D Scenarios

Case	1	2	3	4	5
β (%)	7.5	5.0	8.1	8.7	11.5
β_N	5.7	3.8	6.2	5.8	6.0
I_p (MA)	1.6	1.6	1.6	1.8	2.2
$I_{bootstrap}$	1.07	1.45	1.85	1.92	2.1
I_{ECCD}	0.35	0.50	0.03	0	0
I_{FWCD}	0	0	0	0	0
I_{NBCD}	0.25	0.11	0	0	0
I_{OH}	–0.07	–0.46	–0.28	–0.12	–0.10
q_{95}	6.5	5.0	5.0	5.3	3.6
q_0	3.8	2.3	2.5	3.9	3.4
q_{min}	2.6	3.3	2.1	2.3	
$T_i(0)$ keV	15	12.3	18.5	14.5	19
$T_e(0)$ keV	8.5	9.7	7.0	12.7	13
$n_e(0)$ 10^{20} m^{-3}		0.59	0.89	0.72	0.88
$\bar{n}$ 10^{20} m^{-3}	0.57	0.35	0.54	0.48	0.53
n_{edge} 10^{20} m^{-3}		0.23	0.23	0.23	0.21
$\bar{n}/n_G$	0.4	0.26	0.4	0.32	0.3
P (MW)	20	14	12	14	14
P_{NBI} (MW)	6.5	4.0	4.0	4.0	4.0
P_{EC} (MW)	7.0	6.0	4.0	6.0	6.0
P_{FW} (MW)	6.5	4.0	4.0	4.0	4.0
W (MJ)		1.25	1.3	4.6	6.0
τ_E (s)		0.21	0.29	0.28	0.4
H_{89P}	3.5	3.4	4.4	4.0	4.95
ρ_{*e} at $\bar{T}_e$	2.3×10^{-4}	2.5×10^{-4}	2.1×10^{-4}	2.8×10^{-4}	2.9×10^{-4}
ρ_{*i} at $\bar{T}_i$	1.3×10^{-2}	1.2×10^{-2}	1.5×10^{-2}	1.3×10^{-2}	1.5×10^{-2}
ν_{*e} at $\bar{T}_e$	1.2×10^{-2}	4.2×10^{-3}	8.0×10^{-3}	2.9×10^{-3}	2.1×10^{-3}
ν_{*i} at $\bar{T}_i$	2.7×10^{-3}	1.8×10^{-3}	0.8×10^{-3}	1.6×10^{-3}	0.7×10^{-3}

Case 1: SSC-VH [Turnbull PRL 74, 718 (1995)]
Case 2: β = 5%, P = 16 MW, n & v_ϕ transported
Case 3: β = 8%, P = 15.2 MW, n & v_ϕ transported
Case 4: β = 8%, P = 17 MW
Case 5: β = 11%, P = 17 MW

They all seek steady-state negative central shear current profiles for stability at high β_N. They also seek high bootstrap fractions and make up any difference in the plasma current and the bootstrap current by use of rf current drive. The resulting plasmas have rather low ρ_* and very low ν_*, but they match up well along dimensionless parameter scaling paths to future tokamak devices (Section 2.4 of the Five-Year Plan). The scenarios all have rather low densities and high temperatures. The combination of low density (well below the Greenwald limit) and high power will make it particularly challenging to obtain radiating, detached divertors in these scenarios.

For reference, the first scenario is that published by (Turnbull, 1995, see also St. John IAEA 1994 and Taylor EPS 1994). At that time, DIII–D had seen plasmas with hollow current profiles, very high central betas (calculated to be second stable) and had also seen in other discharges the VH–mode, a transport barrier formed around $\rho = 0.8$. The scenario described considered combining these two features into what was called then Second Stable Core VH–Mode (SSC-VH). The inverted q profile and suitably broad pressure profile was shown through stability calculations to give β_N of 5.7 assuming wall stabilization. Various transport models were used for the electrons including INTOR scaling, the Rebut-Lallia-Watkins model and the Hsieh model for electrons. Essentially the ion diffusivity was taken to be neoclassical near the core (the transport barrier model here was a small multiplier times neoclassical ion transport inside the radius of q_{min}) and rising to 5 times neoclassical near the edge. A combination of bootstrap current which peaked off axis and off-axis ECCD were used to sustain the hollow current profile. On-axis NBCD was used to control the central current density. Fast Wave heating sustained the core electron temperature. A limitation of this scenario was the use of a fixed density profile; no density transport was considered. The rather broad density profile used still contributed a significant bootstrap current. Steady-state solutions were found with the required current profiles and pressure profiles for the high values of β_N in this scenario.

For the Five-Year Plan, we constructed transport simulations using a full but complex model of E×B shear stabilization of turbulence to dynamically form the transport barrier in the simulation. The current profile was evolved to steady state verifying the compatibility of the transport barrier with the second stable core. The model is diagrammed in Fig. 2.3–1 on page 2.3–4 of the Five-Year Program Plan. The model calculates the turbulence shearing rate with no free parameters from the Hahm-Burrell formula based on the evolving density, temperature and rotation speed profiles. Then the local value of $\omega_{E\times B}$ is compared to a model of the turbulence growth rate. This prescription for E×B shear suppression is based on gyrofluid turbulence simulations of Waltz, 1994. The location where the transport barrier forms depends upon both the growth rate profile (which tends to rise from the center) and the source (heating, momentum and fueling) profiles. The barrier usually forms first at the edge due to the high power flux density and the strong density gradient (fueling). The H–mode edge can be suppressed (in the model) by a combination of high edge radiation and low recycling. We did so in order to concentrate on the core transport barrier properties and not on the more difficult to model edge L-H transition. A transport barrier then forms first in the core owing to the peaked heating and/or momentum sources. In the experiments, the reduction of the ITG mode growth rates due to hot ions and fast ion dilution have been found to aid internal transport barrier formation. The negative magnetic shear also eliminates MHD ballooning modes. Raising the power makes the internal transport barrier expand as the E×B shear pushes out against the rising growth rate. This model of transport barrier formation contains many feedback loops, since the radial electric field depends on all the profiles, and a very rich set of anticipated phenomena. It is also hard to run since both the model growth rate and $\omega_{E\times B}$ depend on local gradients.

The cases considered all model an L–mode edge. The transport barrier forms where the turbulence shearing rate from the radial electric field exceeds the local growth rate of the turbulence. When density transport is turned on, the strong local fueling source at the edge easily forms an edge transport barrier which can quickly lead to excessive edge pressure gradients. A large part of the NCS research thrust is aimed at controlling the edge pressure. To avoid this problem, we imposed a large edge growth rate to keep the edge in L–mode and fixed the edge density. The thrust to use an L–mode edge is one of DIII–D main AT thrusts but considering the high power flow through that edge, substantial mantle radiation or other means to suppress the L-H transition will have to be found. These are issues for future experimental and simulation work.

Despite the complexity of the model, the results in Table 5 represent another set of internally consistent numbers of target β_N and H factors with the required power levels and locations of current drive required to produce the necessary current profiles. The target β_N and H factors are large and represent ultimate goals for the DIII–D AT Program. Even with the high H factors, a 10 gyrotron system is needed and the total summed heating and current drive power of 20 MW will be a challenge to the divertor power handling capability in long pulse.

For the near term scenarios, perhaps some of the qualitative features seen in these transport barrier modeling efforts are worth noting. The density transport equation was turned on in Cases 2 and 3 and a transport barrier was allowed to form in the density channel (Fig. 16). Density gradients are more effective than temperature gradients in creating bootstrap current and for that reason, we obtain more bootstrap current than in the original SSC-VH scenario. Also, we have moved the transport barrier further out in radius and that also increases the total bootstrap current. We find it rather easy (in fact too easy in these simulations) to obtain full bootstrap current. It appears that with central fueling from beams or pellets and a longer time for the density to accumulate, we should see strong transport barrier formation in the future through density gradient with accompanying large bootstrap fractions. This research area has only just started on DIII–D. We completed the central Thomson scattering system and it is now operational on DIII–D. With it we will finally be able to see what is happening in the density channel when transport barriers form in the plasma interior. The UCLA group installed an x–mode interferometer on DIII–D about a year ago so we could get an early glimpse of a density transport barrier. One profile (Fig. 17) shows a spectacularly high density gradient, showing us what exciting phenomena may lie ahead in these studies. We have a plan in the 1999 campaign in the Thrust 7 on ITB control to use the inside launch pellet injection together with the counter beam injection to stimulate the formation of transport barriers in the density profile.

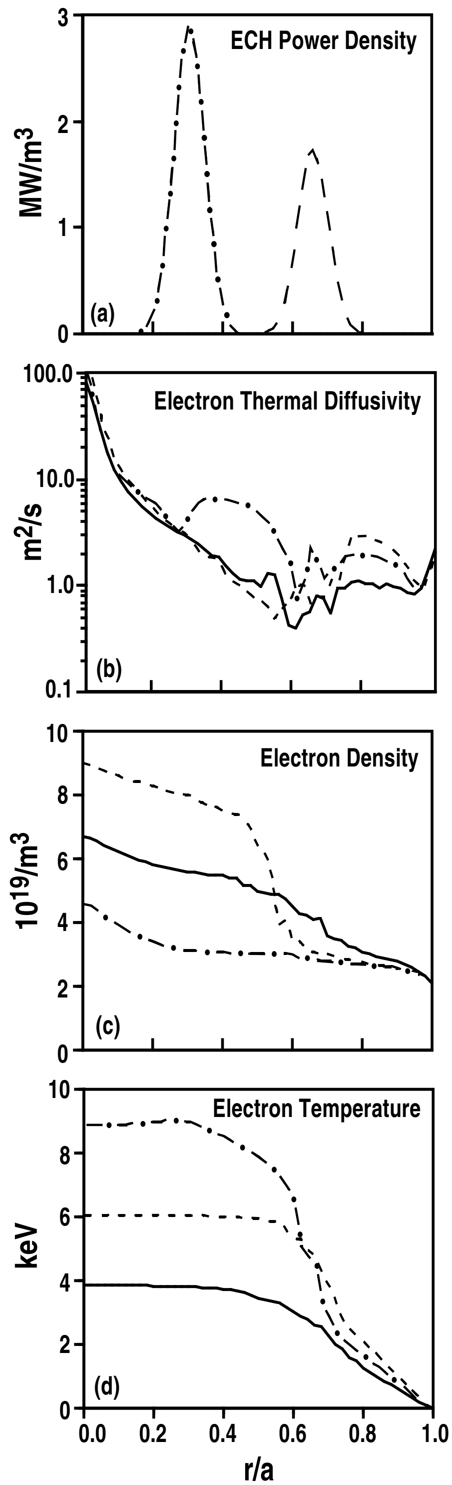

Fig. 16. ECH barrier control is illustrated by a three-case comparison: 6 MW NBI only (solid), 6 MW NBI + 6 MW ECH at r/a ~ 0.7 (dashed), 6 MW NBI + 6 MW ECH at r/a ~ 0.3 (dot dashed). Shown are profiles of (a) ECH power density, (b) model electron thermal diffusivity, (c) electron density, and (d) electron temperature.

Another interesting but not fully understood result was that the ECH was very effective at moving the location of the transport barrier. To see such dynamics was a principal reason for using the complex E×B shear model. The ECH deposition profile is about as narrow as the gradient regions of the transport barrier, and so the ECH is a precision tool for barrier control. We found that ECH applied just outside the radius where a transport barrier was beginning to form would draw the transport barrier out to larger radius. Equally striking but not so positive was the effect of ECH when applied inside a formed transport barrier.

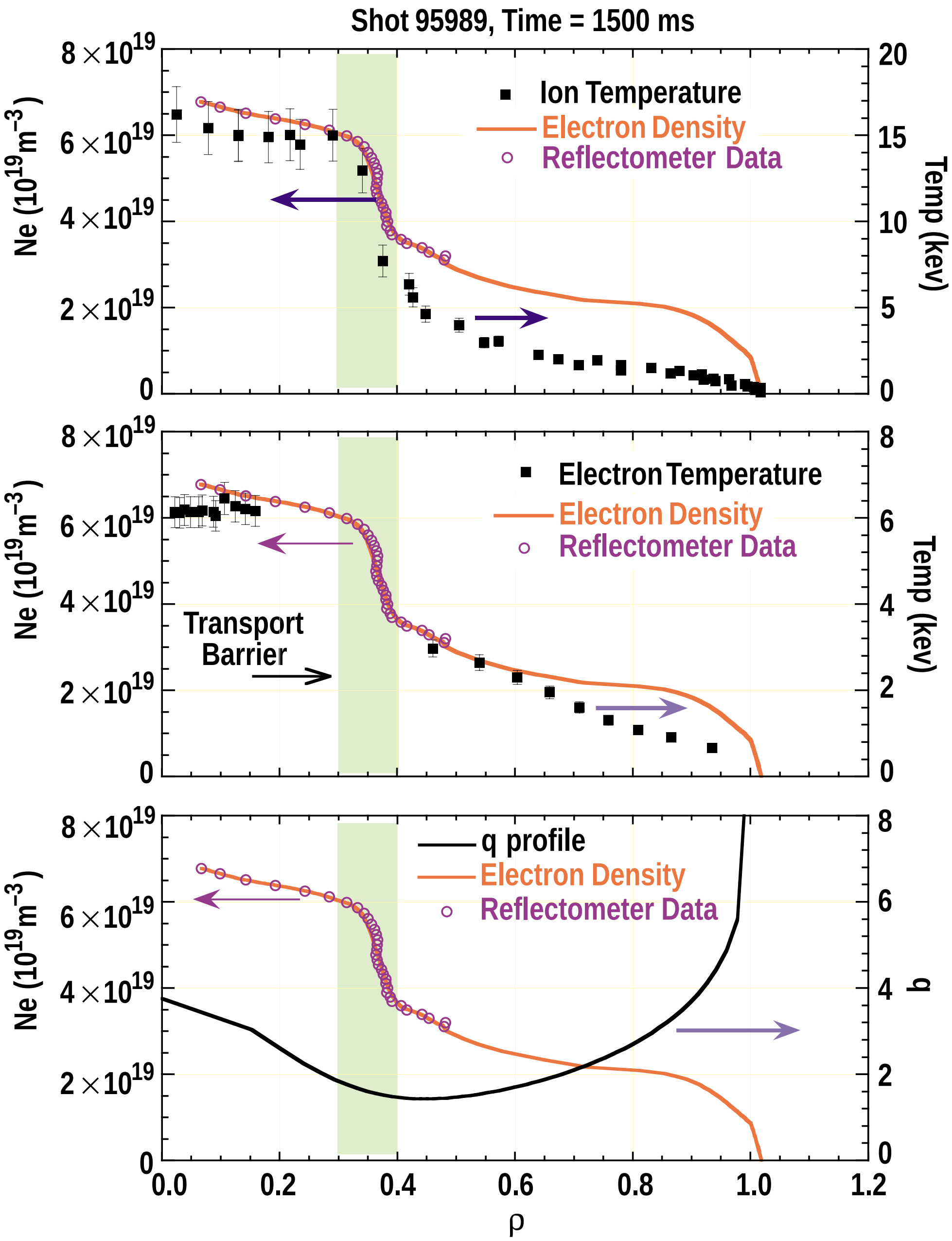

Fig. 17. Transport barrier in the density.

The transport barrier was found to retreat to just inside of the ECH absorption layer. In the model this was due to the fact that the model growth rate increased with the electron temperature gradient but the E×B shear only depends on the ion temperature gradient. Thus, electron heating caused a loss of the E×B shear suppression. A retreat of an existing internal transport barrier with central ECH heating has been observed on DIII–D. Linear growth rate calculations suggest that the excitation of electron temperature gradient modes may be the cause These modes are not expected to be stabilized by E×B shear since they have very large growth rates (and small wavelengths).

The physics picture mentioned above in which $\omega_{E\times B}$ pushes out radially against a radially rising turbulence growth rate suggests that the way to move the transport barrier out in radius is just to add power or momentum inside the formed transport barrier. However, because the transport coefficients are so low inside the barrier, increasing the power, especially localized power, can produce wild swings in local gradients affecting not only $\omega_{E\times B}$ but stability as well. The way through these complications is to increase power slowly. Using this approach, we were able in 1998 to make an internal transport barrier discharge that lasted the whole length of the neutral beam pulse (5 s). This was done at low current where the plasma was beset with Alfvén eigenmodes which had the beneficial effect of throwing out enough fast ions so that the central NBCD was weak and the central q stayed high. When such discharges were attempted at higher current, the Alfvén eigenmodes were eliminated but the central NBCD drove q_0 below one and instabilities terminated the high performance phase.

In the 1999 Thrust 7 campaign, counter-NBI was used as a means to achieve near stationary q profiles with elevated q_0 and to evaluate techniques to expand the ITB. With co-NBI, the sheared E_r from the plasma rotation and that from the pressure gradient are in opposition. This gives a sharp gradient in E_r (large local $\omega_{E\times B}$) but the dynamics are such to make it difficult to increase the radius of the large local $\omega_{E\times B}$. In contrast, with counter-NBI, the E_r from the plasma rotation and the pressure gradient are additive. The local values of $\omega_{E\times B}$ are not as large as for the co-NBI case, but as the beta increases, the region of large $\omega_{E\times B}$ expands and the ITB with it.

The various cases have varying assumptions about how low the transport rates become inside the transport barrier. In DIII–D we have already seen ion neoclassical transport rates all across the cross section so this assumption for the residual transport was made in all cases. But it is clear that similarly low levels for transport rates for electrons and particles in DIII–D are too good. Beta limits would be quickly exceeded. DIII–D does not presently see as much transport reduction in the electron and particle channels as in the ions and apparently will not require it to reach the scenarios shown.

These are some of the interesting phenomena we have seen in our initial exploration of the possibilities for AT physics in the plasma core. The simulations presented give a feeling for the parameter regimes achievable, the power levels in various systems to achieve them, the density and edge control that may be required. But the main value of such simulations is to open a wide vista of new phenomena that should open up as the auxiliary capabilities of DIII–D are developed toward the goal of long pulse sustainment of AT operating modes.

2.2.4. RECENT ARIES-AT SCENARIOS AND THE QUIESCENT DOUBLE BARRIER MODE

We have also explored optimization of AT scenarios as part of the ARIES-AT study. These studies have pushed conceptually and computationally closer to what might be the ultimate stability and transport potential of the tokamak. Here we have looked for pressure profiles consistent with very high bootstrap fractions (f_{BS} = I_{BS}/I_P > 0.9) and high beta values (wall stabilized). Equilibria are found with β_N = 5.6 (wall stabilized), q_{95} = 3.3 and f_{BS} = 0.92, as shown in Fig. 18. The high fraction of bootstrap current at such a low q value is a consequence of the large value of q_{min} ~ 2.5 and the large radius of q_{min}, $\rho_{q_{min}}$ ~ 0.8. The key physics to such a scenario is the ability to form and control a transport barrier at large radius, ρ_{ITB}; with small gradients near the boundary. Analysis shows that sufficient sheared E×B flows to stabilize ITG modes in such a plasma are feasible. Understanding and controlling shorter wavelength micro-instabilities remains a challenge. The ARIES-AT study clearly defines a challenging approach for the optimization of DIII–D NCS plasmas — expanding and controlling the transport barrier at large radius and this effort is the focus of Thrust 7, the Internal Transport Barrier Thrust.

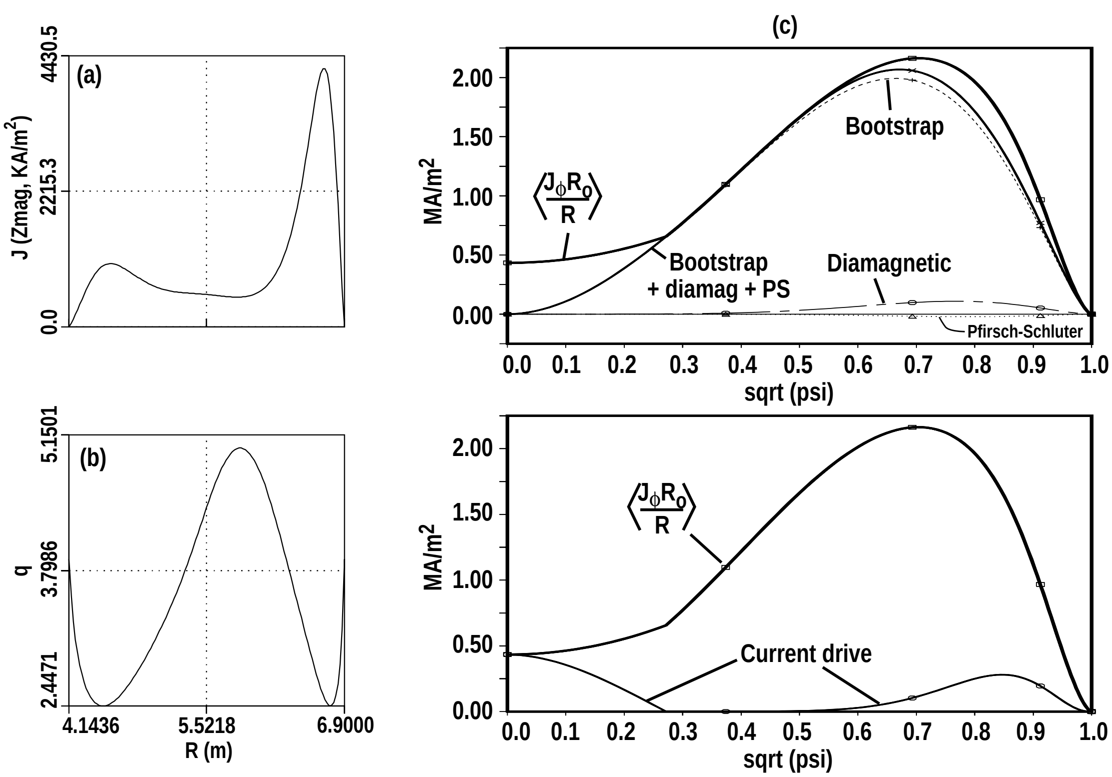

Fig. 18. High bootstrap fraction ARIES-AT scenario: (a) plasma current; (b) q profile; (c) total current (solid), bootstrap (dashed), and diamagnetic current (chain dashed).

The Quiescent Double Barrier Mode

In 2000, the internal transport barrier thrust found a new mode of tokamak operation with great promise toward the advanced scenarios suggested by the ARIES work. This mode had both an internal transport barrier and the edge transport barrier of H–mode (Fig. 19). Despite the edge being in H–mode, the edge plasma is free of the periodic instabilities known as ELMs. The ELMs, a signature feature of H–mode, dump heat and particles in short intense bursts out into the scrape-off layer and these heat and particle loads flow directly to the divertor plates. These ELM heat pulses are projected to be too large and to cause excessive divertor plate erosion in future devices. Hence operation without ELMs is desirable, but all previous operation worldwide in the transient ELM-free H–mode phase immediately following the L-H transition (the only recent exception being the EDA mode in Alcator C–Mod) has shown uncontrolled density rises and accumulation of impurities until either the ELMs began or the discharge suffered a radiative collapse. Hence it was a great surprise that this new mode had no ELM but nevertheless we were able to maintain a steady plasma density with the divertor pumps and the low Z impurity levels stayed steady. There is an issue of high Z impurity concentrations near the discharge center consistent with neoclassical impurity transport with the steep density gradient shown in Fig. 19. There is a continuous MHD instability in the edge of these plasmas, dubbed the edge harmonic oscillation after its distinguishing spectral feature. This continuous MHD activity seems to enable the required density and impurity exhaust. Because of the quiescent nature of the edge of this plasma and the two transport barriers, this mode has been named the quiescent double barrier (QDB) mode.

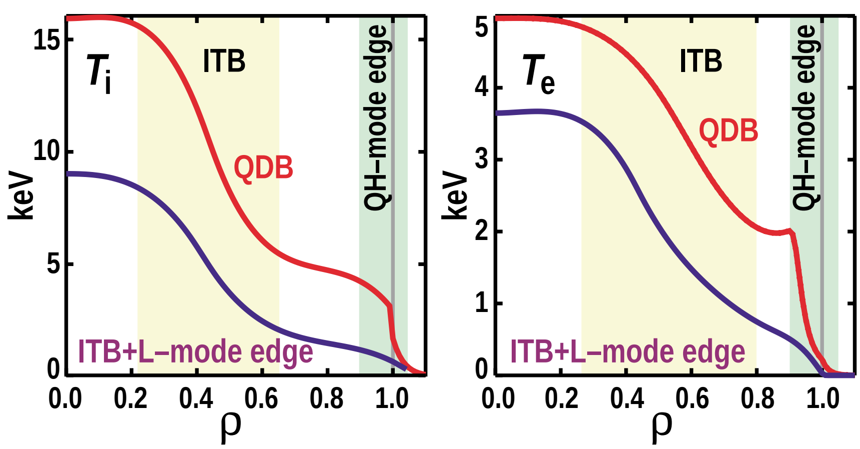

Fig. 19. New operating regime obtained with counter-NBI — quiescent double barrier regime. QDB regime combines: core transport barriers, and quuiescent H–mode (QH–mode) edge barrier.

The profile features of this mode are almost ideal with a view toward the ARIES scenarios. The internal transport barrier is presently located too far into the plasma; it must be moved radially outward to come close to or even merge with the edge H–mode transport barrier. If that can be done, the resulting pressure profiles and bootstrap current profiles will be similar to the ARIES profiles. The QH–mode plasma edge has ideal characteristics for a fusion plasma.

This QDB mode has potential as a high performance steady-state mode. The pressure profiles that might result from moving the internal barrier outward should support very high bootstrap fractions. This mode has lasted several seconds in DIII–D (Fig. 20). The current profile is slowing evolving and amenable to control with ECCD to retain a minimum q value well above 1. The QH–mode edge is ideal for steady-state. The discharge in Fig. 20 reached a β_N*H_{89P} product of 7, nearly the same as the best long duration high bootstrap fraction thrust plasma. Research along this line needs to show that this mode can be obtained in an interestingly wide operating space. The keys to producing this mode are counter neutral beam injection and divertor pumping to hold down the density. The pumping and the low density are consistent with our future ECCD plans. The counter injection brings with it counter current drive, which is not desirable. We need to isolate the physics effect that the counter neutral beam is having. If that effect has to do with creating an edge electric field pattern, we may be able to create that physics effect by other means. This mode may provide motivation for reorienting two neutral beams on

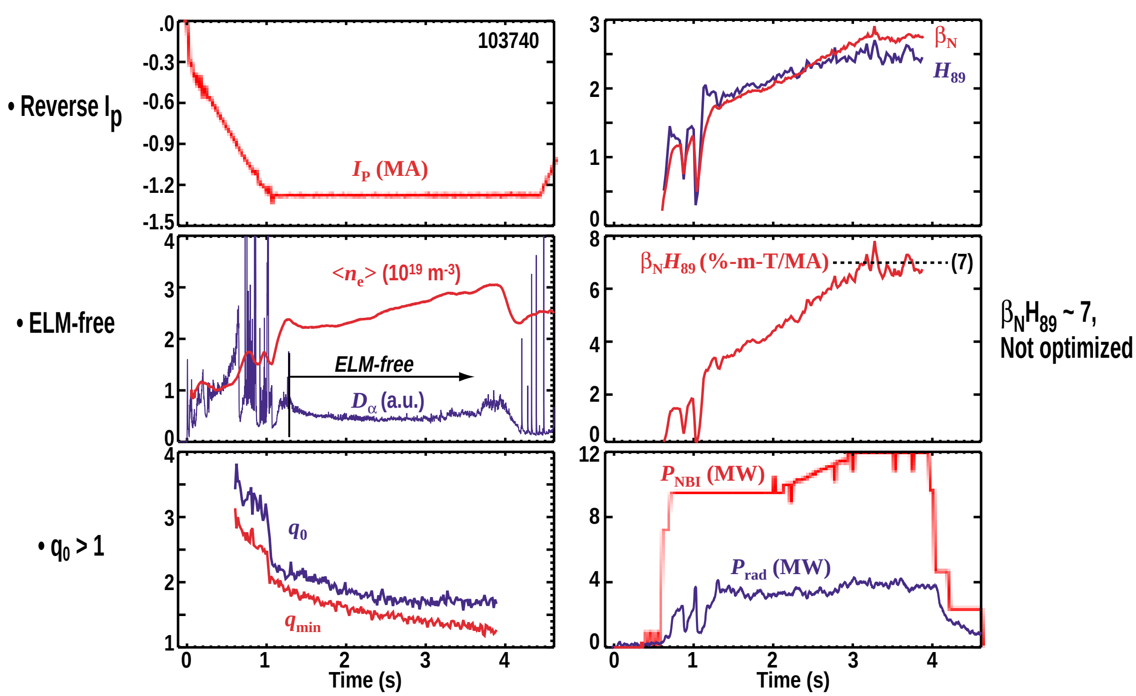

Fig. 20. QDB is a core ITB with a quiescent QH–mode edge, no sawteeth.

DIIID to achieve balanced injection. For future large fusion devices, if the key physics effect is the edge electric field (produced by orbit loss) or edge momentum, then low energy neutral beams just for that purpose might be provided as a relatively minor part of the whole project. At any rate, this QDB mode has such promise that it must be vigorously pursued in the near future to understand how it works.

2.3. THE HIGH INTERNAL INDUCTANCE SCENARIO IN DIII–D

The high internal inductance (high ℓ_i) scenario is the second possible approach to AT performance being pursued in DIII–D research. This scenario is motivated by the well-established experimental observation that both the beta limit and the confinement multiplier increase approximately linearly with ℓ_i. This scenario has the advantages that the current and q profiles are monotonic, requiring less precise tailoring, that the current density and pressure gradient at the plasma edge are not so large as to require wall stabilization to reach $\beta_N \approx 4$, and that the required external current drive would be peaked at the axis which is more efficient and easier to implement than in the NCS scenario.

In order to achieve high values of β_N, relatively broad pressure profiles are required. This places the regions of high pressure gradient toward the discharge edge and, in discharges with large bootstrap current fraction, produces relatively broad current profiles. So, operation at high β_N and high bootstrap fraction tends to lower the self-consistent value of ℓ_i.

The achievable value of β_N is expected to be consistent with the empirical scaling $\mathrm{Max}(\beta_N) \approx 4\ \ell_i$. Thus, a $\beta_N \approx 4$ operating point would have $\ell_i \approx 1$, a larger value than in the NCS scenario but smaller than the maximum values achieved in previous research on ℓ_i scaling. Simulations have shown that bootstrap fraction in the range 50%–70% can be obtained self-consistently with $\ell_i \approx 1$. A sample equilibrium of this class is shown in Fig. 21. With strong shaping ($\delta \geq 0.7$) and flat J and q profiles in the center of the plasma, an optimized equilibrium can be found which is stable to n = 1 ideal modes and to n = ∞ ballooning at $\beta_N = 4$ without a conducting wall. This case has not as yet been examined for transport requirements although the bootstrap current profile is required to be consistent with the assumed pressure profile.

The achievable value of ℓ_i for a given bootstrap fraction can be increased by reducing the value of the safety factor on axis. Reducing q_0 below 1 requires stabilization of the sawtooth instability. Previous work has indicated that sawtooth stabilization is possible with rf heating. A key question for the high ℓ_i scenario is whether rf sawtooth stabilization can be done while maintaining the other requirements (high f_{BS} and β_N). Note that sawtooth stabilization should remove a primary source of perturbations which can initiate neoclassical tearing modes, which in turn may raise the beta limit. An

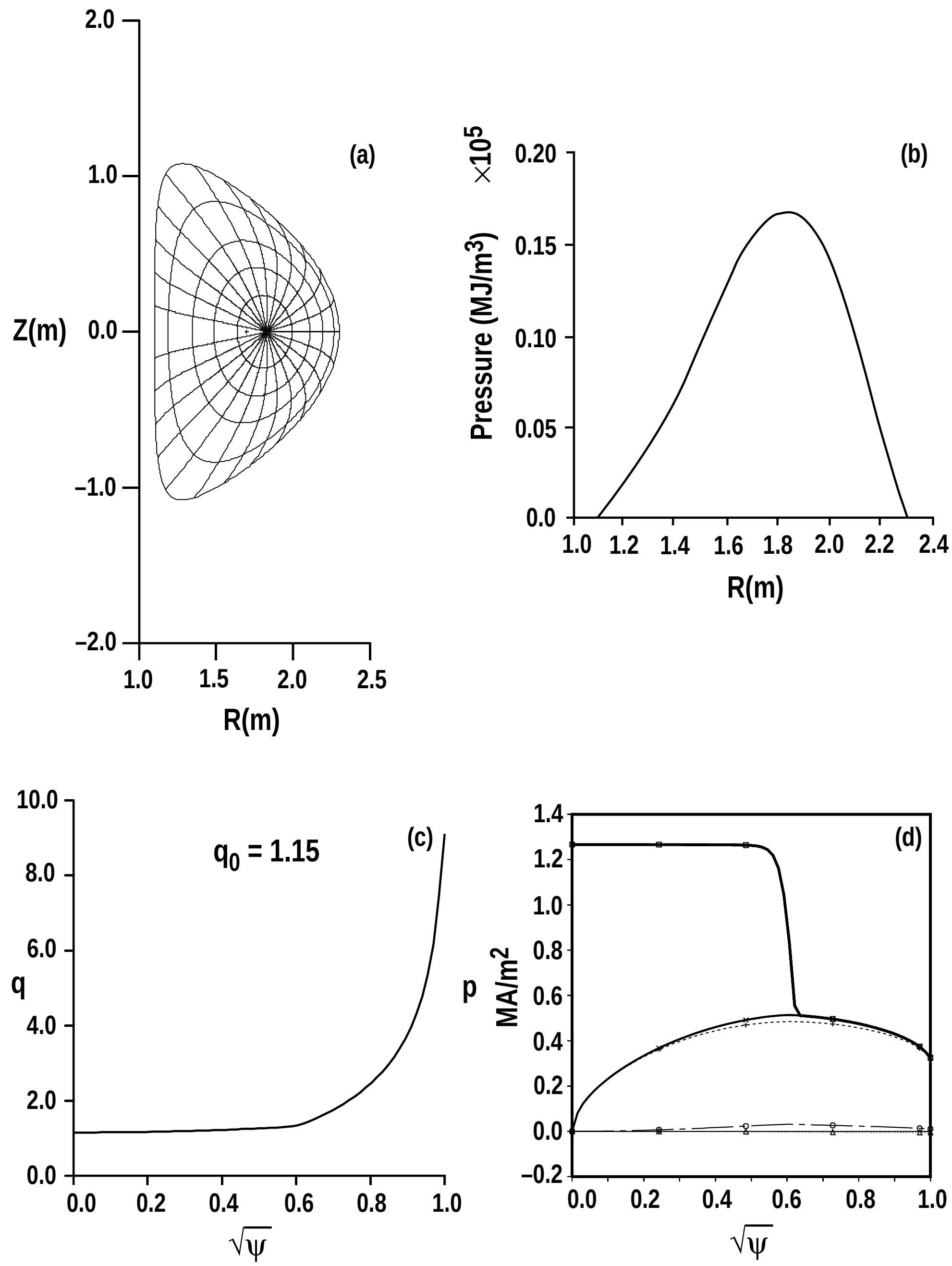

Fig. 21. Optimized ℓ_i, high β_N, edge aligned bootstrap equilibrium in the full-size DIII–D configuration with R = 1.7 m, a = 0.6 m, κ = 1.8, δ = 0.7, B = 1.9 T, I_p = 1.1 MA, q_0 = 1.15, ℓ_i = 0.92, $p_0/\langle p\rangle$ ~ 3.0, β_N = 4.0, and f_{BS} = 70, q_{95} = 6.5. (a) Flux contours, (b) pressure profile across the midplane as a function of major radius, (c) q profile as a function of $\tilde{\psi}^{1/2}$, and (d) flux surface averaged toroidal current densities as function of $\tilde{\psi}^{1/2}$. Here, solid squares represent the total plasma current. The sum of bootstrap (dotted curve), diamagnetic (open circles), and Pfirsch-Schluter (open triangles) contributions is represented by crosses. parison: 6 MW NBI only (solid), 6 MW NBI + 6 MW ECH at r/a ~ 0.7 (dashed), 6 MW NBI + 6 MW ECH at r/a ~ 0.3 (dot dashed). Shown are profiles of (a) ECH power density, (b) model electron thermal diffusivity, (c) electron density, and (d) electron temperature.

example of this scenario is shown in Fig. 22. This example was developed with the same rules as the NCS example cited in Section 2.2.1, i.e., primarily to assess heating and current drive requirements.

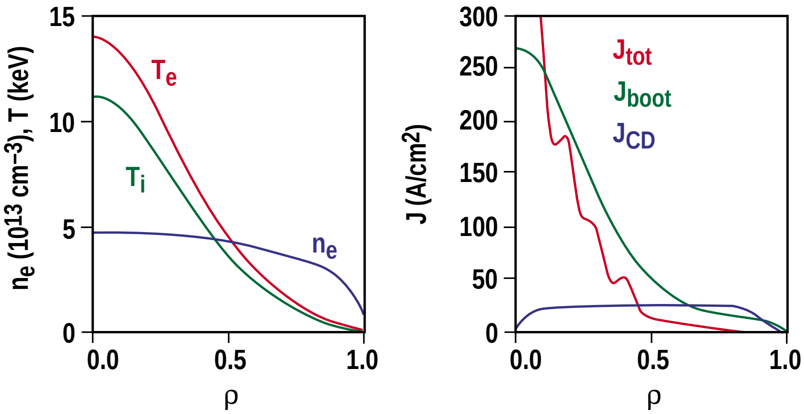

Fig. 22. A high ℓ_i steady-state with $\beta_N H > 10$ can be sustained by a 3 MW ECH system. The parameters are: B = 1.9 T, ℓ_i = 1.6, β_N = 2.5, I_{BS} = 0.41 MA, I = 1.0 MA, β_N = 4.0, P_{FW} = 3.6 MW, I_{FW} = 0.17 MA, q_0 = 0.55, H = 3.0, P_{EC} = 2.4 MW, I_{EC} = 0.14 MA, q_{95} = 5.6, n = 4.2×10^{19} m^{-3}, P_{NB} = 5.0 MW, I_{NB} = 0.27 MA.

Thus there are two distinct versions of the “high ℓ_i” scenario, one requiring some current profile tailoring to maintain q_0 above 1 and sufficiently flat in the central region, and the other requiring effective stabilization of sawteeth.

One criticism of the high ℓ_i scenarios has been that excessive external current drive power would be needed in a reactor. To examine this question explicitly, we have looked at spreadsheet modeling of three high ℓ_i cases and compared them with ARIES-RS. Fixed parameters (with Aries-RS values in parentheses) are R = 5.0 (5.52) m, a = 1.8 (1.38) m, κ = 1.8 (1.7), δ = 0.7 (0.5), q_{95} = 6.5 (3.5), P_{fusion} = 2500 (2167) MW, and $n/n_{Greenwald}$ = 0.95 (1.78). Some of the results are summarized below.

	ℓ_i = 1	ℓ_i = 1.25	ℓ_i = 1.5	ARIES-RS (ℓ_i = 0.42)
β_N	4	5	6	4.84
q_0	1.15	0.85	0.55	2.78
B (T)	7.5	6.58	5.95	7.98
I (MA)	15	13.15	11.9	11.3
f_{bs}	0.61	0.67	0.60	0.88
P_{CD} (MW)	169	126	123	81
H_{89P}	2.1	2.44	2.65	2.35

The benefits of high ℓ_i operation are clear. If satisfactory sawtooth suppression is possible and the increase in β_N can be demonstrated, the q_0 = 0.55 case has significantly lower magnetic field than ARIES-RS, with roughly the same total current. Although the driven current fraction increases by 233%, from 0.12 to 0.40, the required external power increases by only 52%. This is because the current is driven at the axis, where T_e is high and trapping is small making the current drive much more efficient. Further, because the current is driven at the axis, a less complex profile control system is needed.

The multi-year goal of research for the high ℓ_i operating mode is to determine feasible scenarios for steady-state high ℓ_i discharges in DIII–D consistent with the available tokamak resources. A goal would be to maintain elevated ℓ_i values for twice the inductive decay time and confirm that the corresponding increase in confinement and stability is also maintained. The issues to be resolved over several years of work are:

1. Establish whether sawtooth stabilization is both possible and practical.
2. Establish the practical limits to β_N in the two high ℓ_i scenarios without additional wall stabilization. Do the linear relationships between β_N and ℓ_i, and between H and ℓ_i extend to $q_0 < 1$ cases?
3. Establish the current drive requirements for steady-state sustainment of these two scenarios. How much current profile control is needed for the $q_0 > 1$ case?
4. Development of entirely self-consistent scenarios to find the optimum combination of current, density, and temperature profiles.
5. Select the $q_0 > 1$ or the $q_0 < 1$ approach.

Regrettably, due to the intense competition for run time on DIII-D, we do not anticipate being able to allocate run-time to this research thrust until 2002.

To outline the possible content of a future plan to pursue the high ℓ_i scenario, we list here a simplified three-year view of the necessary research:

Goal: $\beta_N \bullet H_{89P} > 10$ with no inductive flux

<u>Year 1</u>

Demonstrate sawtooth stabilization for >1 s and validate the stabilization model. This includes modification and commissioning the ABB transmitters for operation at 60 MHz.

Develop the 3 MW ECH target scenario with $\beta_N \bullet H_{89P} > 10$ transiently. This includes FW coupling studies under the appropriate edge conditions and identification of core pressure limits.

<u>Year 2</u>

Demonstrate the 3 MW ECH integrated scenario. Develop a 6 MW ECH target scenario.

<u>Year 3</u>

Demonstrate the 6 MW ECH integrated scenario. Develop a 10 MW ECH target scenario.

2.4. ADVANCED TOKAMAK RESEARCH THRUSTS 2001

2.4.1. H–MODE PEDESTAL AND ELMS, RESEARCH THRUST 1 (Leader: R.J. Groebner, Deputy: T.H. Osborne)

Three Year Goal

The three year goal of Thrust 1 is to *develop the scientific basis for predicting the height of the H–mode pedestal and ELM effects on the plasma core*. This goal is motivated by the desire of the U.S. magnetic fusion energy community to develop a predictive capability for confinement. It is clear that in H–mode plasmas, the boundary conditions imposed by the H–mode pedestal and by the ELMs must be known in order that core transport models can be reliably used in a predictive fashion. At present, there is no reliable model for these boundary conditions; therefore, pedestal and ELM physics must be considered among the weak links in the ability to predict core confinement.

The goal of Thrust 1 is strongly aligned with several implementation approaches of the recent IPPA report (http://vlt.ucsd.edu/IPPAfinalrev.pdf). These include approaches 3.1.1.2, 3.1.1.3, 3.1.2.1 and 3.1.4.2 which, respectively, are “Understanding Transport Barriers,” “Integrated Models of Core and Edge Physics,” “Understanding Observed Macroscopic Stability Limits” and “Coupling Between Edge and Core Plasmas.” The pursuit of this goal is being made as a thrust due to the fundamental importance of the topic for plasma confinement and due to its inter-disciplinary nature. Studies of the H–mode pedestal and of ELMs requires contributions from the four DIII–D topical science areas: MHD stability, confinement and transport, divertor/edge physics and heating and current drive.

Present Understanding

The present understanding of pedestal and ELM physics is briefly summarized here. The transition to the H–mode state occurs at the very periphery of the plasma and the transition results in rapidly increasing gradients of density, temperature and pressure at the edge. These increases are a signature of markedly improved edge confinement, which is understood to be a result of $E \times B$ shear suppression of turbulence (http://fusion.gat.com/pubs-ext/MISCONF99/Burrell-hmode.pdf). For sufficient heating power, the pressure gradient increases until it reaches a limit set by MHD stability, at which point, an ELM occurs. The ELM instability temporarily but dramatically reduces the edge gradients, removes particles and energy from the plasma and acts as a mechanism to regulate the edge profiles. A comprehensive model, (http://fusion.gat.com/pubs-ext/APS99/Ferron_vgs.pdf) developed within the last two years, shows that the ELM trigger is due to a medium-n ideal kink/ballooning mode. The effects of ELMs are generally observed in the core of the plasma (http://fusion.gat.com/pubs-ext/PSI98/A22860.pdf); rapid perturbations of the electron temperature penetrate into the core and a significant fraction of the energy removed by the ELM comes from the core. The QH–mode, discovered during the last two years, shows that it is possible to maintain large pressure gradients in DIII–D without ELMs. Instead of ELMs, the QH–mode exhibits a continuous MHD mode, called the edge harmonic oscillation (EHO), which removes particles in a smooth and continuous fashion from the plasma, as opposed to the large and intermittent losses caused by ELMs.

Uncertainties

MHD stability at the plasma edge depends sensitively on the details of the edge current density profile. Due to the presence of large edge pressure gradients, it is anticipated that the bootstrap effect produces a very substantial amount of current density at the edge. Direct measurements of the current density are needed both to validate the model for bootstrap current and to allow for more reliable calculations of stability limits. For these purposes, a diagnostic to measure the current density is being installed on DIII–D and will be commissioned in the 2001 campaign. This diagnostic will measure the pitch angle of the magnetic field from Zeeman polarimetry of a lithium beam.

Unknowns

There are several important physics questions, regarding the pedestal and ELMs, about which very little is known. Several of these will be discussed here. These questions motivate the Thrust 1 experimental program for 2001. One of the most fundamental questions for pedestal physics is, "What physics sets the width of the H–mode transport barrier?" The most prevalent hypothesis is that the width is set by the ion poloidal gyroradius, however DIII–D data directly contradict this hypothesis. The most complete transport models that have been developed for the H–mode barrier (Staebler-Hinton, Lebedev-Diamond-Carreras) suggest that the particle flux at the edge sets the scale size. In fact, simple models have been developed to show how neutral fuelling can produce the observed shape of the edge electron density profile. These will be tested this year.

One of the most important questions for ELM physics is: “How does the ELM cause energy and particle loss from the core of the plasma?” A first step to answer this question is to determine how the ELM, which is primarily an edge phenomenon, couples to the core. Some of the ideas for how this may happen include: (1) mode coupling to low order rational q surfaces in the core due to the structure of the eigenfunctions for the ELMs; (2) rapid propagation of the temperature reduction at the plasma edge into the core due to the “stiffness” of the temperature profiles; (3) destabilizing of core MHD modes due to the inward propagation of the edge pressure drop caused by the ELM. The mode coupling idea will be examined this year.

The existence of the QH–mode raises several questions. Some of the first questions to answer may not directly address issues of pedestal formation and ELM physics. Nevertheless, due to the potential for the answers to change conventional ideas of H–mode physics, Thrust 1 will help address some of the basic issues. First, “What are the parameters which control access to the QH–mode?” Thrust 1 will work with Thrust 7 to study this question. The edge harmonic oscillation (EHO) in the QH–mode has the interesting property that it causes increased particle transport with little effect on the energy transport. This property provides control of the plasma density. The question must be answered: “What is the edge harmonic oscillation?” The structure of the EHO will be probed this year in order to examine this question.

2.4.2. THREE-YEAR PLAN AT SCENARIO, RESEARCH THRUST 2 (Leader: M. Wade, Deputies: T. Luce, J. Ferron)

The high q_{min} approach is the primary AT scenario being pursued by DIII–D in its long term development of the AT potential. The mission of this scientific research thrust is to develop an existence proof and the scientific basis for steady-state, high performance tokamak operation. The goals of the AT Scenario Scientific Research Thrust are based firmly in the physics required to realizing this scenario in steady-state. In the AT scenario, steady-state and high performance can only be achieved via the generation of signficant levels of plasma bootstrap current and the maintenance of a hollow current profile using off-axis ECCD to prevent resistive diffusion of the off-axis current peak. For a given q profile (chosen by optimal MHD stability and transport properties) and density profile (chosen for optimal efficiency of the ECCD), the bootstrap current fraction f_{BS} scales proportionately with β_N while the ECCD-driven current I_{ECCD} scales with β_e. Hence, to generate the high f_{BS} and high I_{ECCD} necessary to achieve steady-state in such a scenario, both high normalized performance (β_N ~ 4) and high absolute performance (β ~ 2β_e ~ 5%) are required. Thus, these high levels of performance are not simply lofty goals but are stringent requirements necessary for achieving a steady-state tokamak.

The existence proof that such a scenario exists was developed in 1999. Guided by previous scenario modeling, exploratory experiments to determine the limiting β at parameters suitable for the demonstration (B = 1.6 T, I = 1.2 MA) were carried out and discharges with β_N H_{89} ~ 9 for 2 s were obtained with β_N = 3.8 and β = 4.5% . Analysis of the internal loop voltage in this type of discharge indicates that about 75% of the plasma current is supplied non-inductively, of which ~50% is attributed to bootstrap current. The analysis shows (consistent with the original scenario modeling) that the edge current is consistent with being entirely bootstrap current, the central current is overdriven by the NBI, and the remaining Ohmic current is at the half radius. This implies that replacement of some of the neutral beam (NB) power with off-axis ECCD should lead to a fully non-inductive current sustainment. The limiting process on the attainable β in the quasi-steady phase has been identified to be resistive wall modes which are low frequency (<100 Hz) n = 1 magnetic perturbations that appear just above the no-wall β limit. These plasmas and this strategy at this time essentially sought to produce an integrated AT demonstration within the constraints of free-boundary stable plasmas. The duration of the high performance phase is limited by the resistive evolution of the current density profile to where a resistive wall mode (in some cases combined with a tearing mode) grows and irreversibly ends the high performance phase.

After 1999, new divertor hardware was installed in the upper divertor of DIII–D to allow divertor exhaust of high triangularity (δ ~ 0.7) upper single-null plasmas. This divertor was successful in producing the density control needed to enable sufficient ECCD efficiency for our scenario. Also a reduction of low Z (carbon) impurities by about a factor of two was an added benefit. However, since δ ~ 0.9 in the originally developed scenario, this installation required the use of a plasma shape somewhat different from the optimized shape used in 1999. Achievable values of β_N for longish durations in 2000 were 10%–15% smaller with a shape optimized for pumping than values achieved in the optimized shapes of 1999. Because this reduction in β_N has a significant impact on both the attainable f_{BS} and I_{ECCD}, detailed studies were carried out to quantify the variation of the β limit with plasma shape. The result of this study is shown in Fig. 23 where the maximum β_N is plotted versus the shape parameter S = (I/aB) q_{95}. For reference, the S = 6.7 for the 1999 optimized performance shape while S = 5.2 for the 2000 optimized pumping shape. While the empirical results seem clear, we are still working on whether this apparent shape dependence of the beta limit can be obtained from theory calculations. In 2000, since the divertor was available to provide the density control previously lacking, we

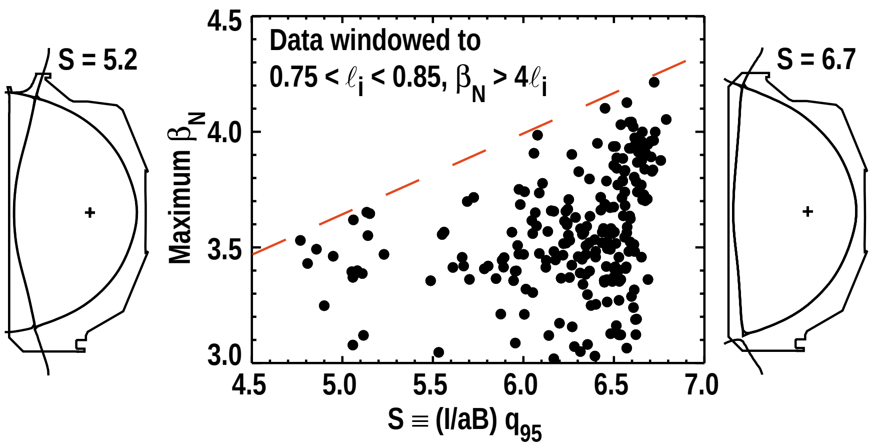

Fig. 23. Measured β_N limit vs. shape parameter. Dataset is restricted to cases with $\beta_N > 4\ \ell_i$ and $0.75 < \ell_i < 0.85$.

sought to find out how long a duration discharge we could make at high performance. The result was a discharge that lasted 6.3 seconds, (35 energy confinement times) with a product β_N*H_{89P} = 7 in ELMing H–mode (Fig. 15). Beta was regulated constant about 5%–10% below the 2/1 tearing mode limit (these discharges had a stationary 3/2 tearing mode present). The density was also regulated constant by gas fueling and divertor pumping. These discharges show the kind of long pulse integrated performance we are seeking to demonstrate, but at higher product β_N, H factor, and bootstrap fraction. The next closest achievement worldwide for this kind of very long duration is product β_N*H_{89P} = 5 in ASDEX-Upgrade. The lower β_N in these plasmas also meant lower bootstrap fraction f_{BS} ~ 40%, pointing out the challenges involved in the integration of seemingly disparate but nevertheless required physics elements (in this case, density control and high β) in achieving steady-state, high performance tokamak plasmas.

The strategy to be used over the next three years in achieving the mission of this scientific research thrust will be to focus on establishing an existence proof, while developing the scientific basis necessary to achieve this goal. In doing this, we will seek to integrate the various physics elements/control tools developed in other portions of the DIII–D program as they become available, seeking to find the optimum “compromise” between these various elements. The main physics integration questions that require immediate resolution in order to clarify the viability of an integrated solution with the present DIII–D hardware are: (1) can β_N ~ 4 be achieved in an optimized pumping configuration? and (2) can ECCD can be used effectively to modify/control the current density profile in a high performance plasma without having deleterious effects on other aspects of the developed scenario? Each of these questions address critical path issues for the development of the AT existence proof on DIII–D. It is envisioned that the resolution of these issues will require detailed systematic experimental studies to understand the basic physics elements involved. In this regard, this thrust will endeavor to explore the underlying reasons why such limitations are apparent and seek means by which to circumvent these limitations.

Upon successful resoluton of these issues, it is envisioned that this research thrust will move aggressively towards demonstrating steady-state, high performance operation. This is consistent with the expected buildup of EC power on DIII–D over the next three years, increasing steadily from the present system to an eight-gyrotron system. More importantly, the five newest gyrotrons will be equipped with diamond windows to enable longer (10 s) pulses. Subsequent optimization of normalized and absolute performance will take advantage of ongoing hardware upgrades and physics understanding advances in the areas of resistive wall mode stabilization, neoclassical tearing mode stabilization, and turbulence-driven transport. While it has always been clear that wall stabilization is a necessary element of the path to Advanced Tokamak plasmas, the RWM research line has grown in importance since it appears now more difficult to achieve sufficiently high β_N within the constraints of free boundary. Our strategy will focus on bringing the resistive wall mode stabilization along more rapidly to use it to gain access to higher β_N states. We will also further investigate the role of plasma shape in achievable β_N with a view toward a possible future divertor modification. Since “surgical” control of the current density profile is required in this scenario, ECCD is considered an integral aspect of the overall solution and demonstration of its ability to modify the current profile is of utmost importance.

2.4.3. NEOCLASSICAL TEARING MODE, RESEARCH THRUST 3 (Task Leaders: R.J. La Haye and C.C. Petty)

Thrust 3 on neoclassical tearing modes is to advance the physics understanding of NTMs, including the means of stabilization. Increased understanding of the science of thresholds allows predictive capability and knowledge of how to avoid them. Development of tools for stabilization, if NTMs are unavoidable, includes radially localized off-axis ECCD (PoP done), sub-resonant static helical fields (needs PoP) and modulated radially localized off-axis ECRH (needs PoP).

Two principal research lines are foreseen in a three-year plan with efforts on threshold physics and stabilization in each which can apply to each line: (1) studies in H–mode with sawteeth present (the mainline candidate for a burning plasma device) and (2) studies in an AT mode with raised q_{min}. The three-year plan is shown in Fig. 24 with accomplishments for the years 1999 and 2000 from the Thrust 3 inception.

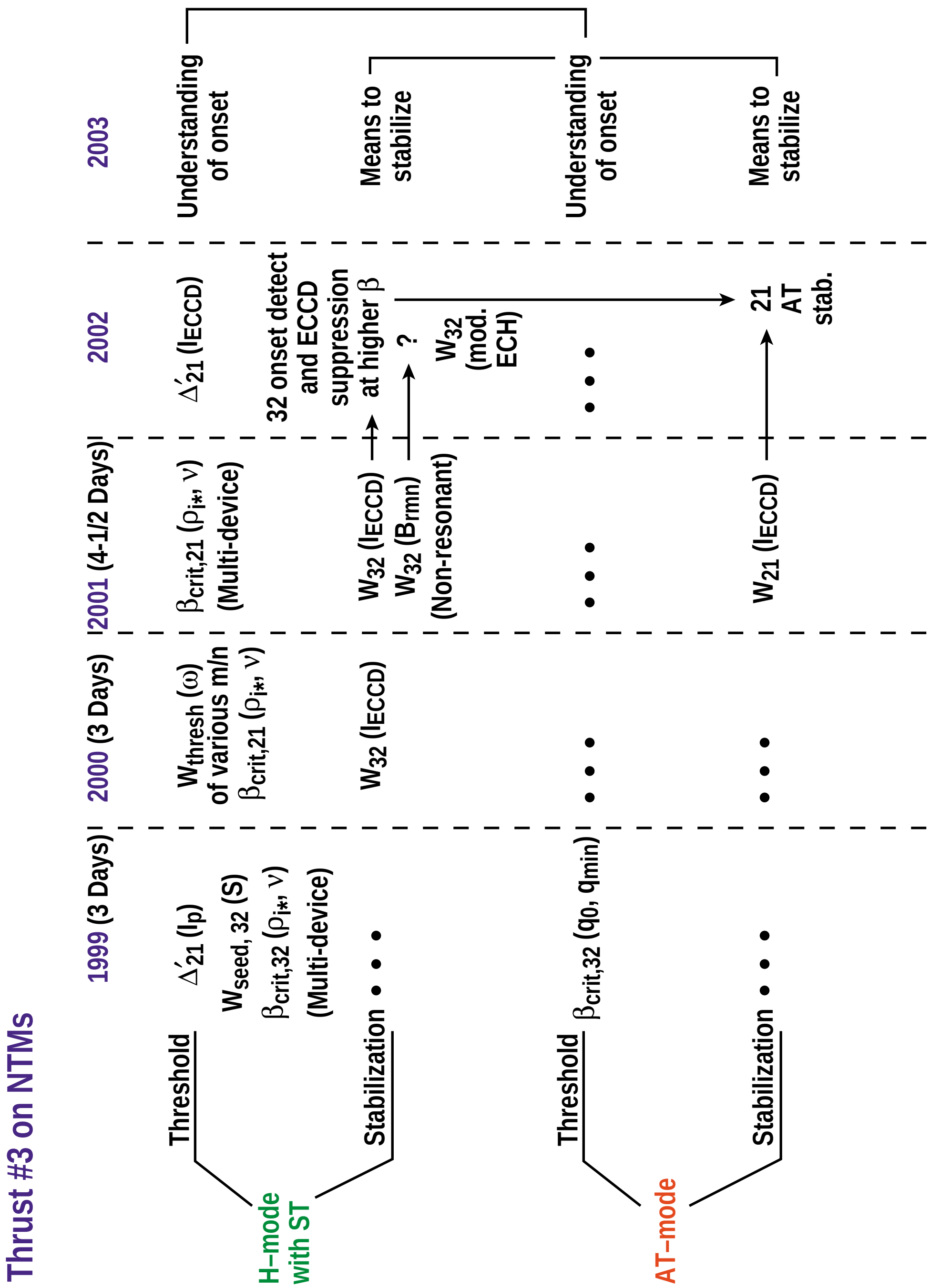

Fig. 24. NTM Research Plan

Papers have been published or are being prepared for submission on: (1) classical tearing mode $\Delta' > 0$ in an ohmic plasma, (2) multi-device scaling of critical beta versus dimensionless parameters including the scaling of seed islands with magnetic Reynold's number, (3) raising the critical beta for an NTM by control of the q-profile and (4) island propagation in the guiding center frame.

H–Mode With Sawteeth

<u>2001</u>

Finish multi-device scaling of m/n=2/1 mode critical beta

Finish ECCD stabilization of m/n=3/2 mode, including PCS active control of the optimum suppression and stabilization of the m/n=3/2 mode in presence of "frequency coupled" sawteeth instabilities.

Proof of principle of stabilization of m/n=3/2 mode by sub-resonant, static externally applied helical field.

<u>2002</u>

Continue classical tearing mode studies with ohmic plasma <u>destabilized</u> by rad. loc. off-axis ECCD.

PCS onset detection and prompt ECCD suppression of m/n=3/2 NTM so as to raise beta limit without mode.

Proof of principle of modulated, rad. loc. off-axis ECRH suppression of m/n=3/2 NTM.

<u>2003</u>

Understanding of Δ', W_{thresh}, W_{seed} for prediction of NTM onset

Best means of routine stabilization.

AT Mode with q_{min} > 1

<u>2001</u>

First attempt at m/n=2/1 NTM suppression by rad. loc. off-axis ECCD.

Tool development in H–mode line for suppression.

<u>2002</u>

Continued tool development in H–mode line for suppression.

Application of ECCD, ECRH, and/or helical field for suppression of m/n=2/1 NTM in $q_{min} \leq 2$ AT discharge.

<u>2003</u>

Understanding of Δ', NTM onset with q-profile.

Best means of routine stabilization.

2.4.4. RESISTIVE WALL MODE STABILIZATION, RESEARCH THRUST 4 (Leader: A.M. Garofalo, Deputy: L.C. Johnson)

The resistive wall mode (RWM) is a global, long-wavelength MHD instability that grows in the plasma on the relatively long time scale of the magnetic flux penetration through the resistive wall, τ_W. In tokamaks, as well as in several other magnetic fusion confinement devices, the RWM has been predicted to limit the maximum plasma performance (in terms of the plasma normalized beta) to less than a half the performance achievable with a conducting wall [Turnbull, 1995; Kessel, 1994; Miller, 1997].

The objective of the DIII–D Research Thrust #4 is to advance the physics understanding of RWM stability, including the dependence on plasma rotation, wall/plasma distance, and active feedback control, with the ultimate goal of achieving sustained operation at beta values close to the ideal-wall beta limit through passive or active stabilization of the RWM.

RWMs have been identified in RFPs [Alper, 1990; Greene, 1993] and in tokamaks [Strait, 1995; Ivers, 1996; Okabayashi, 1996; Garofalo, 1999; Takeji, 2000]. Since the initial observations in DIII–D of the effectiveness of wall stabilization, high-performance, wall stabilized plasmas have been produced transiently which satisfy the requirements for attractive high-β_N steady-state tokamak operation [Strait, 1995; Garofalo, 1999]. However, in agreement with MHD predictions, the onset of the RWM has been observed to be the limiting factor in the lifetime and β_N achieved in DIII–D advanced tokamak regimes with low internal inductance [Petty, 2000; Garofalo, 2000]. Two schemes to control the unstable RWM have been proposed: (1) the plasma rotation with respect to the walls can be maintained (e.g. with neutral beam injection) [Bondeson, 1994]; and (2) a network of active feedback coils can be configured so as to simulate: (a) a perfectly conducting wall [Bishop, 1989; Jensen, 1997], or (b) a "fake" rotating wall [Fitzpatrick, 1996], or (c) to provide direct feedback on the RWM amplitude [Boozer, 1998; Okabayashi, 1998].

In DIII–D the plasma rotation is observed to slow down when $\beta_N > \beta_N^{\text{no wall}}$ resulting in the failure of passive rotational stabilization [Garofalo, 1999]. Suppression of the RWM by active feedback control using the existing DIII–D six-element error field correction coil (C–coil) has been clearly demonstrated [Garofalo, 2000; Okabayashi, 2000]. Development of both 1-D analytic lumped element [Jensen, 1999] and 3-D numerical finite element models of feedback [Bialek, 2000] have shown very good agreement with measurements of feedback. These models have been used both to optimize present experiments and to plan for significant improvement in feedback effectiveness in future experiments approaching the ideal wall β_N limit.

Progress in 2000: Investigate Plasma Rotation Slowdown, Validate Models for Active Control and Optimize Control With Six-Element C-coil

RWM and Plasma Rotation

- Factor of 2 increase in error field correction currents improve plasma performance and duration near RWM stability limit.
- Discovery of error field amplification by marginally stable RWM, suggested by theory, provides new mechanism for rotation slowdown above no-wall beta limit.
- Improved capabilities for measurement of RWM structure.
  - Tested and debugged new arrays of external saddle loops.
- Mode detection on SXR and Mirnov data consistent with n=1 ideal kink

RWM Feedback Modeling

- 1-D feedback simulation code predicts accurately feedback system dynamics.
- GATO-VACUUM 2-D codes (which include self-consistently the effect of feedback on plasma eigenfuction) predict little effect of feedback on mode structure.
- VALEN 3-D code used for extensive optimization studies of feedback configurations.
  - Predicts large gain in stable beta with internal RWM sensors.
- Successful initial benchmarking between feedback models

RWM feedback experiments

- Clear demonstration of RWM suppression by feedback for >20 τ_W at β_N-$\beta_N^{\text{no wall}}$ exceeding feedback limit predicted by VALEN (plasma rotation effect).
- Complete RWM suppression observed at moderate $\beta_N \geq \beta_N^{\text{no wall}}$
  - Weakly unstable plasma created by slow plasma current ramp.
  - Feedback control stabilizes RWM for more than 500 ms (to end of I_p ramp).
- Observation of rigid RWM structure when feedback is applied.

- Comparison of different algorithms.
  - Improved Smart Shell algorithm by using independent C–coil pairs.
  - Optimization of feedback parameters using 1-D model agrees with experimental results.

2001: Establish Limits of Passive Wall Stabilization, Benchmark Quantitatively Feedback Models and Optimize Control With Improved Sensors and Six-Element C-Coil

- Validate model of plasma rotation slowdown.
- Operate RWM feedback using new sensors and compare performance with different sensors and with theory (benchmark VALEN)
- Demonstrate improved steady state beta limit in high performance AT plasmas through feedback stabilization of the RWM.
- Finalize design of upgraded active coil.

2002-2003: Feedback With Upgraded C–Coil and Additional Amplifiers

- Install and operate upgraded active coil.
- Complete experimental description of parametric dependence of the RWM stability.
- Routine use of RWM feedback for steady state operation significantly above the no-wall beta limit.

References:

B. Alper, Phys. Fluids **B2**, 1338 (1990).
J. Bialek, APS-DPP Invited Paper 2000 and Phys. Plasmas to be published.
C. M. Bishop, Plasma Phys. and Control. Fusion **31**, 1179 (1989).
A. Bondeson and D. J. Ward, Phys. Rev. Lett. **72**, 2709 (1994).
A. Boozer, Phys. Plasmas **5**, 3350 (1998).
R. Fitzpatrick and T. H. Jensen, Phys. Plasmas **3**, 2641 (1996).
P. Greene and S. Robertson, Plasma Phys. **5**, 556 (1993).
A. M. Garofalo, A. D. Turnbull, E. J. Strait, et al., Phys. Plasmas **6**, 1893 (1999).
A. M. Garofalo, A. D. Turnbull, M. E. Austin, et al., Phys. Rev. Lett. **82**, 3811 (1999).
A. M. Garofalo, E. J. Strait, J. M. Bialek, et al., Nucl. Fusion **40**, 1491 (2000).

A. M. Garofalo, et al., IAEA 2000 and Nucl. Fusion to be published.
T. H. Ivers, E. Eisner, A. Garofalo, et al., Phys. Plasmas **3**, 1926 (1996).
T. H. Jensen and R. Fitzpatrick, Phys. Plasmas **4**, 2997 (1997).
T. H. Jensen and A. M. Garofalo, Phys. Plasmas **6**, 2757 (1999).
C. Kessel, J. Manickam, G. Rewoldt, T. M. Tang, Phys. Rev. Lett. **72**, 1212 (1994).
R. L. Miller, Y. R. Lin-Liu, A. D. Turnbull, et al., Phys. Plasmas **4**, 1062 (1997).
M. Okabayashi et al, Nuclear Fusion **36**, 1167 (1996).
M. Okabayashi, et al. Nucl. Fusion **38**, 1607 (1998).
M. Okabayashi, APS-DPP Invited Paper 2000 and Phys. Plasmas to be published.
C.C. Petty, et al., 27th EPS Conference on Controlled Fusion and Plasma Physics, Budapest, Hungary, 2000.
E. J. Strait, T. S. Taylor, A. D. Turnbull, et al., Phys. Rev. Lett. **74**, 2483 (1995).
E. J. Strait, L. L. Lao, M. E. Mauel, et al., Phys. Rev. Lett. **75**, 4421 (1995).
S. Takeji, et al., IAEA Fusion Energy Conference 2000, Sorrento, Italy, Paper EX7/01.
A. D. Turnbull, T. S. Taylor, Y. R. Lin-Liu, H. St John, Phys. Rev. Lett. **74**, 718 (1995).

2.4.5. INTERNAL TRANSPORT BARRIER CONTROL, RESEARCH THRUST 7 (Leader: E.J. Doyle; Deputy: C.M. Greenfield)

Control of internal transport barriers (ITB) was identified as a high priority at the 1999 AT Workshop held at General Atomics as well as at the Snowmass meeting. Specifically, control of three different aspects of the ITB is needed in order to establish it as an element of a viable confinement regime.

First, the spatial extent of the barrier must be extended in order to increase the energy content and the fusion output from within the barrier region (Fig. 25). The requirement that this be done in a manner consistent with MHD stability leads to a second area of control: the pressure gradient in the barrier itself must be maintained at a level below MHD stability limits. Broadening of the bootstrap current profile would also result from success in the first two areas. This in turn would help attack the third requirement, that the current profile must be maintained so that $q_{\rm min}$ remains elevated, with weak or negative shear in the core.

Experiments in 2000 resulted in the discovery of the quiescent double barrier regime, or QDB [C.M. Greenfield, et al., GA-A23595, submitted to Phys. Rev. Lett.]. This regime combines a strong ITB with an ELM-free edge characterized by the EHO. The EHO, which will be jointly studied by Thrust 1 and 7, allows particle control without the core interference normally associated with ELMs. Separation between the core and edge boundaries is maintained by a null in the E¥B shearing rate. This allows an ITB to form

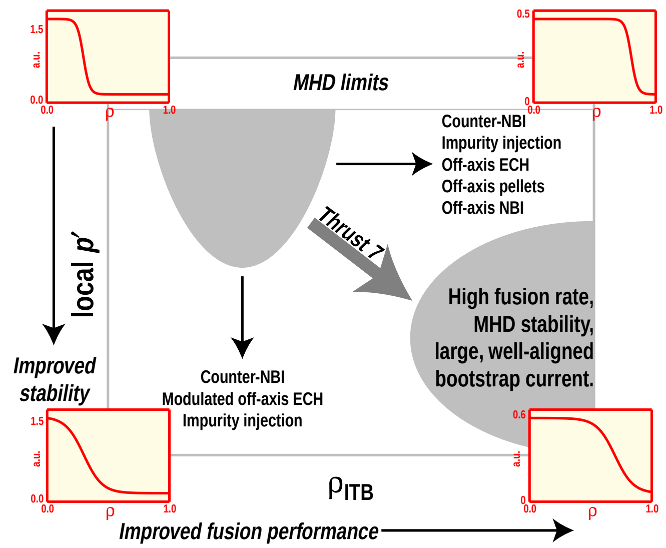

Fig. 25. Efforts in Thrust 7 are focused toward increasing fusion performance by expanding the transport barrier, while at the same time controlling the pressure gradient within the transport barrier to maintain MHD stability.

and with transport characteristics nearly identical to the core of a 1999 L–mode edge discharge, but elevated by the H–mode edge pedestal. In addition, this regime can be sustained for the length of the neutral beam heating pulse.

Further efforts in Thrust 7 will continue development of the transport control tools shown in Fig. 25. These tools are expected to contribute not only to future ITB development efforts, but also to the more general problem of transport control in arbitrary confinement regimes (Fig. 26). In addition, the QDB regime will be studied to determine its potential as a prototype ITB-based Advanced Tokamak regime.

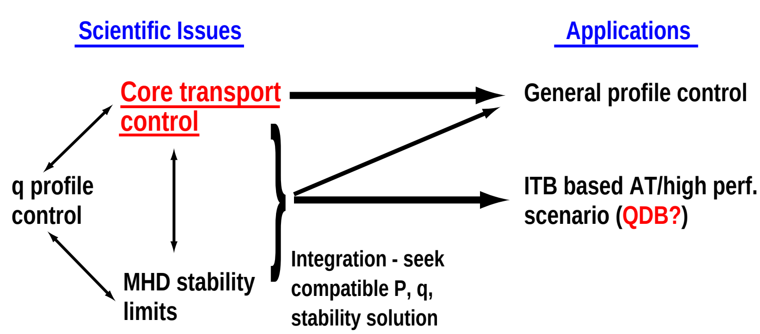

Fig. 26. Efforts in Thrust 7-combine development of general transport control tools with studies of specific candidate AT regimes based on the ITB. The QDB is currently being assessed as a potential candidate regime.

Efforts in 2001 will continue to perform open-loop tests of the tools shown in Fig. 25. An important consideration here is that each of these tools has a foundation in theoretical calculations: Impurity injection is known to result in reductions to the growth rates of turbulence that can increase transport. Off-axis ECH has been predicted to be capable of moving the barrier toward the heating location, or, through use of modulation, moderating the gradients in the barrier. Pellets can both locally reduce the temperature and increase gradients, thereby impacting both the growth rates and the suppression mechanism (E×B shear) for plasma turbulence. Counter-NBI has been shown to broaden the ITB by removing the competition between rotation and pressure gradient effects.

Progress in this area was limited in 2000 due to our retargeting of efforts toward initial evaluation of the QDB. An exception to this was the successful application of impurity injection to broaden a preexisting ITB [J.C. DeBoo, *et al.*, APS 2000]. The experimental plan for 2001 will include open-loop tests utilizing neon and argon injection, pellets and ECH. In 2002 and 2003, it is hoped that we will be in a position to begin closed-loop tests using one or more of these tools. This will require development of real-time calculations of barrier characteristics (primarily location) and adaptation of the plasma control system to control the associated hardware.

A second, related area under study in Thrust 7 is the QDB regime. Initial studies of the QDB regime were very promising, but there are still several questions that must be answered to determine its viability as an AT regime in its own right.

First, all QDB discharges to date have been run at I_p = 1.3 MA and B_T = 1.8-2.0 T. We consider the potential of transferability to other tokamaks to be a critical factor in determining the AT relevance. Thus, parameter scans (current, field, density, power, shape, etc.) will be executed to “map out” the parameter space of the QDB. It is expected that these scans will also provide a large amount of data to further our understanding of the underlying physics of the QDB regime. In particular, the importance of the EHO edge motivates us to carry these experiments out in conjunction with Thrust 1.

Second, the QDB regime appears at first glance to be at a disadvantage with regard to noninductive sustainment. This is partially due to the fact that it has been obtained only with counter-NBI, thus with counter-NBCD. It is believed that this can be attacked through a combination of broadening of the bootstrap current profile and off-axis ECCD. Although this viewpoint is supported by preliminary CORSICA modeling, we will not have enough ECCD available in 2001 for a full effort in this area (only one launcher is available for co-ECCD in these reversed I_p plasmas). A dedicated ECCD test is anticipated in 2002. We will address broadening of the bootstrap profile starting in 2001 with attempts to modify, and hopefully broaden, the exceptionally peaked electron density profile.

A decision point is anticipated for the end of the 2001 campaign on whether the QDB regime should be treated as a serious candidate AT regime. However, it is also seen as an attractive target for some of the tool development efforts, since it does maintain a steady barrier for several seconds. As appropriate, much of the tool development in 2001 is planned to be carried out in the QDB regime. If the decision is made to pursue the QDB as an AT regime, it is anticipated that it will be used as a target for future integration of the various control mechanisms.

3. TOPICAL SCIENCE AREAS — THREE YEAR VIEWS

3.1. STABILITY TOPICAL AREA GOALS (3 YEAR VIEW)

Stability represents a fundamental limitation to the operation of high energy, magnetically confined plasmas. MHD theory (particularly ideal MHD) has been quite successful in predicting and explaining stability boundaries, but many challenges remain, particularly in non-ideal MHD stability. These challenges include the scientific understanding of plasma stability, as well as the practical application of this understanding toward avoiding or controlling plasma instabilities. Advances in the understanding of non-ideal MHD stability should be applicable to devices other than tokamaks, and perhaps even to magnetized plasmas outside of fusion energy research (astrophysical plasmas, for example). Thus, the long-range goal of the stability and disruption physics research in DIII–D is <u>to establish the scientific basis for understanding and predicting limits to macroscopic stability of toroidal plasmas</u>.

Over the next three years, specific goals are to significantly advance the physics understanding in the following areas:

1. Resistive wall mode stability, including the dependence on plasma rotation, wall distance, and feedback stabilization.

2. Edge-driven instabilities in plasmas with a large (H–mode) edge pressure gradient and associated bootstrap current, including means of reducing their impact on the core plasma.

3. Neoclassically driven tearing modes, including threshold mechanisms and means of active stabilization.

4. Non-ideal plasma instabilities such as sawteeth, resistive interchange modes, and fast ion driven instabilities, including stability thresholds and dynamics of the instability.

5. Disruption dynamics and methods of disruption mitigation.

The DIII–D Stability and Disruption Physics program is aimed at addressing a wide range of stability science issues. This broad program of stability science research includes critical issues for Advanced Tokamak (AT) plasmas, and the long-range goals include the understanding and improvement of stability limits in high-performance plasmas with a large bootstrap fraction. With recent advances in scientific understanding and technical tools, we are beginning to study plasmas compatible with steady-state operation at high beta, and to develop active means of controlling stability. While much of the research that is specifically motivated for understanding and improving the stability of AT plasmas tends to be carried out within the DIII–D research thrusts rather than the topical science area, it too advances in our understanding of MHD stability. These advances in understanding and control techniques supported by theory and modeling, will be applicable to configurations beyond the tokamak.

3.1.1. PROGRESS IN 2000

In neoclassical tearing mode physics, detailed measurements of the plasma and island rotation rates provided evidence to support theories of the polarization current as the mechanism responsible for threshold behavior. Complete suppression of the m/n=3/2 neoclassical tearing mode by electron cyclotron current drive was demonstrated, using localized current drive at the island location. Fine tuning of the second harmonic resonance location led to complete suppression of the tearing mode, accompanied by a rise in beta and a 25% increase in neutron emission which persisted after the ECCD was turned off. TORAY code modeling of the location and magnitude of the current drive is in good agreement with theoretical expectations and with the measured island location and width.

Significant advances were made in both understanding and control of the resistive wall mode. The first direct measurements were made of the amplification of an externally applied resonant field by a plasma that is marginally stable to the resistive wall mode; error field amplification is a hypothesis which may explain the rotational slowing that is observed at high beta. Improvements in control algorithms led to feedback stabilization of a weakly unstable resistive wall mode for intervals up to 500 milliseconds. Improvements in magnetic diagnostics led to confirmation of the prediction that the mode structure does not change significantly during feedback stabilization.

A brief experiment using a massive gas puff to induce a thermal quench showed a prompt drop in the core electron temperature, indicating a rapid penetration of the injected gas when the gas flow is sufficiently large. The time history of the density rise was consistent with the time of flight for neutral gas atoms and was similar for different injected species, suggesting that the gas penetrates directly by self-shielding of the gas cloud and not by instability-driven plasma transport. These results are promising for high-pressure gas injection as a disruption mitigation technique.

3.2. CONFINEMENT AND TRANSPORT TOPICAL AREA GOALS (3 Year View)

Significant results were obtained in the confinement and transport topical science area which made major contributions to our EPS, IAEA and APS DPP presentations. Perhaps the most impressive set of results was obtained in the detailed physics experiment on the quiescent H–mode conducted in the core transport physics working group. These results were reported in an invited talk by K.H. Burrell at the APS DPP meeting. First observed in 1999, the quiescent H–mode is an ELM-free H–mode with constant density and radiated power levels. This is very different from standard, ELM-free H–mode where density and radiated power increase continually. An edge MHD oscillation, the edge harmonic oscillation, is responsible for the enhanced edge1 the constant density levels. This oscillation affects the edge particle transport but not the edge thermal transport; edge gradients are the same or slightly steeper in the quiescent H–mode than in the preceeding ELMing H–mode. The quiescent H–mode phase can operate for long periods of time; we have achieved 3.5 second or about 25 energy confinement times, limited only by the plasma current and beam pulse lengths chosen. The conditions required for quiescent H–mode are neutral beam injection opposite to the plasma current (counter-injection) at power levels above 3.7 MW plus cyropumping to reduce the edge plasma and neutral density.

Detailed measurements this year show that the edge harmonic oscillation is an electromagnetic oscillation; there are current, density and temperature fluctuations associated with this oscillation. Scans of the edge plasma location relative to the beam emissions spectroscopy (BES) system show that the density fluctuation peaks on the separatrix and extends out into the open field lines. Oscillations on the open field lines with the same harmonic structure have been seen with Langmuir probes in the scrape-off layer and on the divertor plates.

In addition to the quiescent H–mode studies done in this topical science area, the quiescent H–mode was a key portion of the quiescent double barrier (QDB) discharges which were discovered and investigated during experiments conducted by the internal transport barrier thrust (Thrust 7).

Another important result from the core physics working group was an extensive series of experiments on the use of impurity injection to improve core confinement (RI–mode); these results were reported at the IAEA meeting by M. Murakami. External impurity injection into L–mode edge discharges in DIII–D has produced clear increases in confinement (factor of 2 in energy confinement and neutraon emission), reduction in all transport channels (particularly ion thermal diffusivity to the neoclassical level), and simultaneous reduction of long-wavelength turbulence. Turbulence suppression and transport reduction are attributed to synergistic effects of impurity-induced enhancement of E×B shearing rate and reduction of toroidal drift wave turbulence. A prompt reduction of density fluctuations and local transprot at the beginning of impurity injection appears to result from an increased gradient of toroidal rotation enhancing the E×B shearing. Transport simulations carried out using the National Transport Code Collaboration Demonstration Code with a gyroLandau fluid mode GLF23 indicate E×B shearing suppression is the dominant transport suppression mechanism.

A key experiment was carried out by the nondimensional scaling working group on the ρ_* scaling of turbulence; this was presented at the IAEA meeting by G.R. McKee. Plasma turbulence characteristics, including radial correlation lengths, decorrelation times, amplitude profile and flow velocity, have been measured during a ρ_* scan on DIII–D while all other transport-relevant dimensionless quantities (β, ν_*, κ, q, $T_e/T_{i.}$) are held nearly constant. The turbulence is measured by examining the correlation properties of the local long-wavelength ($k_\perp \rho_i \leq 1$) density fluctuations, measured with beam emission spectroscopy. The radial correlation length of the turbulence, L_c, r, is shown to scale with the local ion gyroradius, L_c, r, $\cong 5\ \rho_i$, while the decorrelation times scale with the local scousting velocity as $\tau_c \sim a/c_s$. The turbulent diffusivity parameter, D ~ (L_c, r^2/τ_c), scales in a roughly gyroBohm-like fashion, as predicted by the gyrokinetic equations governing transport. The experimental one-fluid power balance heat diffusivity scaling and that from GLF23 modeling compare reasonably well. However, the power balance ion thermal diffusivity is found to scale much differently than gyroBohm, indicating that there is still a puzzle to be understood.

In the H–mode physics area, the highlight was further work on the effects of the direction of the ion ∇B drift on the L to H transition. A summary of the work in this area over the past two years was presented by T.N. Carlstrom at the EPS meeting. The novel result from this year's work was the direct observation of change in shear in the poloidal fluctuation propagation speed with the direction of the ∇B drift. These measurements were independently made in L–mode plasmas using beam emission spectroscopy and correlation reflectomtery in plasmas that were vertically shifted up or down to change the location of the divertor X-point thus altering the direction of the ∇B drift relative to the X-point. At a point 5 cm inside the separatrix, the poloidal fluctuation velocity is always in the ion diamagnetic drift direction, independent of the X-point location. Near the separatrix, however, the sign of the velocity depends on the direction of ∇B drift relative to the X-point. For the case of ∇B drift toward the X-point, the velocity is in the electron diamagnetic drift direction. Accordingly, there is a velocity shear layer near the separatrix in the case where the power threshold is lower. This is qualitatively consistent with ideas of shear suppression of turbulence. Interestingly, the E×B drift velocity at the outer midplane separatrix shows little if any change with the ∇B drift direction in this experiment, suggesting that additional factors are governing the poloidal velocity of the turbulence. One possibility is changes in the parallel flow related to changes in the X-point location.

The major experiment in the fundamental turbulence area was a search for more evidence that avalanche-like events are a significant component of radial energy transport. This is the first dedicated experiment for this work although some work was done piggyback last year. Preliminary analysis of the electron cyclotron emission measurements shows that there are intermittent, radially propagating structures seen in the electron temperature. Further analysis is in progress to quantify how much of the energy transport is carried by these events.

Space limitations make it impossible to summarize all 11 days of experiments in this fiscal year in the confinement and transport topical science area. A more complete summary can be found at http://fusion.gat.com/exp/2001/pdf/burrell.pdf.

Physics research aims at a level of understanding that allows quantitative predictions. In the transport arena in magnetic fusion research, this means the development of a predictive understanding of the energy, particle and momentum transport in magnetized plasmas. Achieving this goals requires the combined efforts of theorists, modelers and experimentalists to develop the fundamental theories, include them in numerical models, compare those models with the results of experiments and then iteratively improve them. Based on this ultimate goal, our three year goals for confinement and transport are

1. Develop improved physics understanding and control of reduced core transport regions
   - Develop and exploit new tools for controlling core transport: pellet injection, impurity injection, counter neutral beam injection, co- and counter-ECCD.
   - Broaden tests of the $\Omega_{E\times B}$ versus Γ_{MAX} comparison by using new tools to investigate effect of T_i/T_e ratio, impurities, density peaking, magnetic shear, alpha (Shafranov shift) stabilization
   - Increase emphasis on understanding electron transport and angular momentum transport
2. Investigate fundamental nature of turbulent transport in tokamaks
   - Can we identify features in the data which are unique to the fundamental theoretical microturbulence modes (e.g. ITG, ETG, TEM)?
   - Compare measured turbulence characteristics with gyrokinetic and gyrofluid code predictions.
   - Test predictions about zonal flows.

3. Carry out innovative experiments to make quantitative tests of predictions of (theory-based) transport models .

4. Utilize nondimensional scaling approach to further elucidate tokamak transport.
   – Intermachine comparisons.

5. Test theories of edge and divertor conditions needed to get H–mode.
   – Encourage detailed comparison of edge modelling (e.g. Drake, Xu) with experimental results.
   – Determine if plasma parameters alone govern threshold or whether atomic physics (e.g. neutrals) is also important.

6. Investigate fundamental nature of L to H transition.
   – ∇B drift effect on edge shear.
   – Effect of SOL flow on transition.
   – Effect of divertor geometry and edge poloidal asymmetries on transition.

7. Study H–mode edge pedestal and investigate key physics controlling edge gradient and pedestal values.

3.3. DIVERTOR/EDGE PHYSICS TOPICAL AREA GOALS (3 Year View)

The main function of the boundary plasma is to control particle and power flux at the interface between the core plasma and the material walls. The long range goal of the DIII–D divertor and scrape-off layer science program is to improve our predictive capability of the particle and power control methods. This task involves both experiments using state-of-the art 2-D diagnostics to identify the relevant physical processes, and computational models of the relevant physical processes. In the next three years, we have several specific goals:

- Use physics understanding and modeling to extend the radiative divertor solution to lower core density.
- Better understand SOL and divertor transport, especially drifts.
- Test turbulence and transport theories in double null plasmas near magnetic balance.
- Understand the physics of high density and high confinement operation, this includes the physics of the edge pedestal.

- Understand Plasma Materials Interactions (PMI) issues, such as carbon sources, lithium, and first-wall configurations.

We have previously identified and studied the radiative divertor or “detached” mode of operation which reduces the heat and particle flux in the divertor by deuterium puffing. Normally, detached operation is obtained above a core density that is 60% of the Greenwald density. In DIII–D, intrinsic carbon radiation in the divertor is a key feature of detached operation. In the future, we plan to extend the detached operating regime to a core density of about 30%–40% of the Greewald density, which is appropriate for near-term DIII-D AT operation. This will be accomplished by concentrating radiation in the divertor with injected impurities such as argon. The two tools to achieve this goal are so-called “puff and pump” techniques, (deuterium injection and pumping to provide a force on impurities towards the divertor) and divertor baffling (to better control neutrals).

3.3.1. 2000 DIVERTOR/EDGE PROGRESS

In 2000, the experiments in the edge and divertor area were executed both in the divertor topical science area and Thrust 8. The topical science area had scheduled five run days, and used one day of contingency to complete the experiments. The topics were: impurity sources (1 day), plasma materials interactions (1 day), high density with good confinement (1 day), and high density with the upper divertor (2 days). The work in Thrust 8 was focused on commissioning the new upper divertor hardware for AT operation and included 8 days.

In short, the Thrust 8 experiments demonstrated that the new upper divertor — with two cryopumps arranged in a high-triangularity divertor configuration — could supply adequate density and impurity control for DIII–D AT plasmas. The plasma control system was able to locate the inner and outer divertor strike points at the entrance to the divertor pumps, and densities of 30%–40% of the Greenwald density were achieved. We observed that the particle exhaust in the upper divertor depended sensitively on the up/down magnetic balance of the double-null plasma. An unbalance of only 1 cm (the distance between the two nulls measured at the midplane) was sufficient to increase the exhaust to the upper divertor. New measurements of exhaust efficiency were made possible by a large array of Langmuir probes at the pump entrance. The efficiency is defined as the exhaust rate of the pump divided by the incident ion flux measured by the probes. These measurements were found to be consistent with a neutrals model due to Maingi that was used to guide the design of the upper divertor. Preliminary puff and pump experiments with trace impurities indicated good entrainment of neon and argon at these core densitities.

An additional modification of the upper divertor in FY2000 was the careful contouring of the tiles in the inner strike point region. Previously, the tiles were a series of straight-line approximations to a circle, and thus had leading edges. We had evidence that these edges were heating and could be causing additional carbon influx. With the new contoured tiles, we observed a reduction in the core carbon content for AT plasma operation of nearly a factor of two. In addition, long-pulse AT discharges, with nearly 50 MJ of injected power, were obtained without significant increases in core carbon content. The tile temperature measured by IRTV diagnostics was about 1200°C at the end of the discharge, which is less than the temperature where we would expect graphite to sublimate (~1500°C). This gives us good confidence that this divertor design is adequate for AT operation on DIII–D.

In the divertor topical science area, new insight into carbon transport and sources was made possible by the DiMES probe and spectroscopic measurements. The DiMES probe shows that the sputtering yield of the DIII–D carbon tiles has been reduced by over a factor of 10 due to the nearly 30 boronizations during a seven year period. In turn, we find a long-term change in low ionization states of carbon and the carbon molecular spectra (i.e. CD bands) from the divertor region, showing a reduction of nearly four in the carbon source. However, the core carbon content has remained roughly constant during this period, suggesting that there may be other important sources of carbon such as the main chamber wall. This is consistent with the observation that in detached plasmas, we see no net erosion of the divertor, and often observe net redeposition. These experiments will be carried out in the 2001 campaign, and involve collaborative work with the C–Mod device. Other DiMES exposures included lithium, where it was shown that MHD forces on the lithium must be carefully taken into account.

On C–Mod, they observe large convective losses from the SOL of the main plasma, along with recycling light in the main chamber. These effects are observed to become very important to particle balance during high density operation on C–Mod. Data from the midplane plunging probe on DIII–D also exhibit effects of convective transport in the SOL at high density. We plan to investigate parameter regimes on DIII–D in which we might and might not expect to see the effect of large SOL cross field transport. These experiments will be carried out in the 2001 campaign, and involve collaborative work with the C–Mod device.

The divertor topical science area is also the area where high density, high confinement experiments and pedestal physics studies are carried out. Densities up to 40% above the Greewald density with H_{89} ~ 1.9 were achieved; albeit mostly at a low power of 3 MW. Previously, we felt that pumping was important, but we have now reproduced these results without pumping. The core density profile in these cases has moderate peaking. Most importantly, the properties of the pedestal plasma do not degrade as the density is increased.

An invited paper at the 2000 APS meeting by M.J. Schaffer summarizes the recent work in drifts near the X-point. Detailed measurements by probes and Thomson scattering reveal local electric potential and electron pressure hills near the divertor X-point in DIII–D L–mode plasmas, which give rise to a large E×B circulation of particles, energy and momentum across the separatrix. The potential hill extends from the X-point into closed magnetic surfaces (confinement surfaces),inner and outer scrape-off layers, and private surfaces (between the divertor strike points). Thus, E×B circulates plasma around the X-point, including in and out across the separatrix. The circulation is an order of magnitude more effective at removing momentum than is charge exchange with neutral atoms. The private region convection measured in DIII–D confirms a numerical prediction by Rognlien et al.,who also showed that this flow is responsible for the toroidal field direction sensitivity of divertor plasmas. The coupled electron pressure and potential hills are explained as consequences of an ion temperature gradient (high upstream, low near the X-point) along magnetic flux tubes stronger than the electron temperature gradient. The low ion temperature at the X-point may be sustained by the same E×B circulation which connects cold ions from the divertor target plasma to the X-point. The electron pressure hill appears to be absent in H–mode plasmas, thus indicating absence also of the potential hill and X-point circulation. We speculate that suppression of the circulation and its transport, by uniformization of the ion temperature on the confinement surfaces, might be important for spontaneous transition from L– to H–mode.

3.3.2. EXPERIMENTAL PLANNING

To organize our 2001 experimental campaigns, we formed four working groups:

1. *Extending radiative divertor operation to AT plasma conditions* (lower core density). These experiments involve argon and neon puffing with deuterium puffing at a core density of about 40% of the Greenwald density. (Trace impurity experiments have been carried out in the new upper closed divertor, but radiative levels have not been done.) Any experiments dealing with plasma shape and it's interaction with the divertor plasma (and configuration) will be carried out in this area.

2. *High density at high confinement.* These experiments will try to expand the operational regime of the high density, good confinement discharges. (i.e. higher power). Several pedestal studies are planned as part of these experiments, and there may be inside pellet launch high density operation.

3. *Cross field transport in the SOL.* This work is focused on fundamental transport studies in the SOL for both L–mode and H–mode, along with comparisons with models.

4. *First wall physics and PMI*. The carbon source work will be done in this area. The DiMES erosion experiments, along with other materials such as lithium will be carried out.

2001 Run Campaign

In 2001, we will have several new divertor diagnostics capabilities, including a new upper divertor UV TV (a collaborative effort with Hampton, University), improvements to several visible spectroscopic measurements, and measurements of CV temperature and flow velocity (through a collaboration with Padua, Italy). In addition, several of the computational modeling tools have been modified and improved, particularly the calculation of drifts in the UEDGE fluid code, onion skin modeling (a transport code like ONETWO-CORSICA for the SOL), and BOUT (2D-edge turbulent transport predictions).

The details of the 2001 campaign are presented in Section 4.2.3.

Overview of the 2002 Experiments

We would expect that significant progress would be made in the areas of understanding the plasma flow in the 2001 campaign. In addition, experiments in this area and in other groups (e.g. Thrust 2) will examine more closely the coupling between the core plasma shape and the upper divertor. In this year, we anticipate:

1. Develop more radiative divertor discharges that are relevant for AT (lower density) scenarios. The AT radiative divertor discharges may include radiative mantle experiments.

2. We expect a continuation of studies of the new physics of far SOL transport we will begin in 2001.

3. We would anticipate that UEDGE modeling with drifts would be almost routine. Detailed benchmarking of experimental results with the code output would take place.

4. We would continue to study the role of carbon transport and carbon sources in the AT and high-density plasmas. The relative role of divertor and core plasma

contributions to the recycling and impurity influx would be an ongoing study. DiMES studies of divertor and main chamber erosion will continue.

Overview of the 2003 Experiments

1. Continue to develop radiative divertor and mantle solutions compatible with the density operation needed for the (near-term) AT core plasma scenarios.

2. Attempt an integrated divertor/core demonstration as an element of the NCS scenario work.

3. Make more detailed comparisons of the plasma flow with UEDGE. If necessary, we might propose a new diagnostic to increase our understanding of the details of the flow patterns. Improved modeling of divertor/SOL transport and radiation from higher Z impurities along with carbon would benefit these studies.

The outcome of the years 2002 and 2003 AT and divertor experiments will guide us towards the future course of the divertor effort. The options in the near future are:

1. Accept the single-null/biased double-null configuration, perhaps with a number of refinements such an more contoured tiles.

2. Use the SN, baffled divertor, but change it’s shape slightly as prescribed by the AT scenario experiments.

3. Proceed to a full double-null divertor configuration.

4. Develop fueling scenarios that provide acceptable decoupling and independent control of the core and divertor densities and radiative power dissipation.

3.4. HEATING AND CURRENT DRIVE TOPICAL SCIENCE AREA GOALS (3 Year View)

The primary focus of this topical science area is the development of a predictive understanding of the basic wave/particle physics involved in heating and current drive. The end goal is validation of computational models which interpret the physics under relevant physical conditions and geometry to provide information needed to apply heating and current drive power effectively. This goal requires iterative steps of experiment, theory, and modeling.

The three-year goals in this topical science area are reprioritized from previous years:

1. Establish predictive capability for electron cyclotron current drive (ECCD), including dependencies on density, temperature, geometry, power, trapping, and dc electric field.

2. Develop long pulse discharges with full noninductive current, including discharges with very high bootstrap fraction.

3. Advance the physics understanding of neutral beam current drive (NBCD).

4. Advance the physics understanding of fast wave current drive (FWCD) and heating in the ion cyclotron range of frequencies (ICRF).

5. Develop routine electron heating using the ICRF system.

The emphasis in the immediate future will be on Goals 1 and 2, since the DIII–D AT program is focused on development of discharges with stationary current profiles supported by a combination of ECCD and high bootstrap fraction. Some analysis of discharges with NBCD can be performed, but experiments to explicitly address NBCD issues are not planned in the 2001 campaign due to more pressing issues. Experiments using power in the ICRF are also deferred due primarily to operational issues while the ECH system is being constructed.

3.4.1. PROGRESS IN 2000

Excellent progress was made in the 2000 campaign on ECCD physics. New this year was a pair of ECH wave launchers, the P1999 antenna designed and constructed by PPPL. These launchers are the first on DIII–D to have the ability to steer the beam continuously in both the poloidal (vertical) and toroidal (horizontal) directions. This capability strongly advanced our ability to carry out studies of the physics of ECCD by providing independent control of the $n_{||}$ (parallel index of refraction) and the current drive location. In the past, we had only one degree of freedom, the poloidal steering angle, so changes in the $n_{||}$ and the steering location were linked. With the ability to vary these quantities orthogonally, more scientific studies of the physics of ECCD were enabled.

In addition to new experiments, new tools were developed for improved analysis of ECCD experiments. A simulation technique using MSE signals directly was developed and routinely used to provide more accurate analysis when the driven current profile is very narrow. At the same time, improvements to the set of basis functions used in the EFIT code to model the current profile were made. Limitations on the old set of functions (splines or polynomials) were found to reduce the resolution to a level below that needed for ECCD. Using the new tools, the agreement between the experiment and theory was greatly improved in most cases. This was the subject of an invited talk at the APS/DPP Meeting in Quebec.

Before science experiments could be performed, it was necessary to validate the wave launching conditions for new gyrotrons and the new antennas. An extensive campaign carried out over several days but totaling 2 days yielded full commissioning of the two new Gycom gyrotrons acquired from the Tokamak de Varennes program as well as the P1999 launcher and connecting waveguide in several combinations. Conclusions were that the polarization calculations for every gyrotron/antenna combination were at least nearly correct, good enough for most applications. The poloidal and toroidal steering of the P1999 antenna was also shown to be consistent with calibration data from the test stand.

The first physics experiments were designed to make connection with previous ECCD results. In L–mode, scans were performed as follows: vertical scan of the CD location at fixed $n_{||} = 0.4$; scan along a flux surface to vary the magnetic well depth at fixed minor radius of 0.4 and 0.3 at fixed $n_{||} = 0.4$; and a scan of $n_{||}$ from –0.4 to 0.4 at minor radius of 0.3 and fixed poloidal location (inboard and top). Also, in H–mode a radial scan of minor radius at fixed $n_{||}$=0.4 and an n|| scan at minor radius of 0.2 were performed. Analysis of these experiments is time consuming and not yet completed. Nevertheless, it is clear that co- and counter-current drive can be performed using the P1999 launcher, and well localized current drive has been clearly identified in the H–mode discharges. The profiles of driven current for these many cases will be compared to results from the TORAY-GA ray tracing code and the CQL3D Fokker-Planck code.

Other experiments using the ECH system include the avalanche experiment, where ECH was used to provide control of the electron temperature; the generation of strong electron transport barriers; and the full stabilization of neoclassical tearing modes by localized ECCD. The latter experiment showed indirectly that a large driven current density consistent with theory could be obtained even in high density H–mode discharges with vigorous ELMs, sawteeth, and tearing modes present.

4. SYNOPSIS OF THE 2001 DIII–D RESEARCH PLAN

The research campaign for 2001 has been organized into six research thrusts and a broader selection of experiments in four Topical Science Areas. Significant blocks of experimental time have been allocated to the research thrusts, since these activities are aimed directly at critical objectives for the DIII–D Program and for the tokamak research program generally. Additional experimental time in the topical areas maintains the breadth and scientific depth of the DIII–D Program. Below we convey the essential content of the various research thrust and topical science experiments and their goals and anticipated and hoped for results. The research described has been allocated to 60 run days out of a possible 75 run days, with 15 days of contingency. Additional detailed information can be found on the Web, and related links: http://fusion.gat.com/exp/2001/

The experiment plan was put together with input and prioritization by the year 2001 Research Council. Based on the "DIII–D Five-Year Program Plan 1999–2003," August 1998, GA–A22950, the Research Council develops a three-year plan which is annually updated. The first of these Three Year Plans was made in 1999. Progress on the research thrusts and topical areas in the 2000 experiment campaign was reviewed at the Year End Review (http://fusion.gat.com/exp/2001/review.shtml, also broadcast on the internet) September 7–8, 2000. With input from that review and considering the three-year objectives, year 2001 research thrusts were identified. A call for ideas towards those objectives was issued and over 200 ideas were presented at a community-wide Research Opportunities Forum on November 8-10, 2000 which was broadcast on the internet. Several proposals were presented remotely, including presentations from PPPL and MIT. The various thrust and topical area groups prioritized, combined, and otherwise sifted these ideas. The plans so arrived at were presented to the Research Council in December and the advice of the Research Council was used to set the final allocations of run time for the year 2001 campaign.

The 2001 experiment plan, summarized in Table 6, consists of efforts in six thrust areas and four topical areas. Thrust 1, which did not receive any run-time allocation last year (edge stability), has been slightly rescoped to focus on the edge pedestal. Thrust 5 was a data analysis task last year, and has been completed. Thrust 8 was successfully completed last year, and the operation of the new upper divertor pump and baffle was demonstrated in AT plasmas. Thrust 9, ECH/ECCD validation, was nearly completed, and the remaining tasks will be handled this year in the Heating and Current Drive

Table 6
Run Time Allocations for the 2001 Experiment Campaign

No.	Acronym	Description	75 Day Plan	Area Leader
T1	Edge pedestal	Understand what determines the structure of the edge pedestal in H-mode and the edge localized modes	4.5	<u>R. Groebner (GA)</u> T. Osborne (GA)
T2	AT scenario	Develop the existence proof and the scientific basis for future exploration of high performance, steady-state Advanced Tokamak operation.	8	<u>M. Wade (ORNL)</u> T. Luce (GA) J. Ferron (GA)
T3	NTM	Advance the physics understanding of neoclassical tearing modes, including the thresholds and means of stabilization.	4.5	R. LaHaye (GA) C. Petty (GA)
T4	RWM	Advance the physics understanding of RWM stability, including the dependence on plasma rotation, wall/plasma distance, and active feedback stabilization.	9	A. Garofalo (Col.) L. Johnson (PPPL)
T6	High ℓ_i	Exploration of the high li AT plasma scenario		Deferred
T7	ITB	Develop the ability to create and sustain optimized pressure profiles that are simultaneously consistent with high performance, improved stability, and high bootstrap fraction.	6	E. Doyle (UCLA) C. Greenfield (GA)
		Thrust totals	32	
		Stability topical area	4	T. Strait (GA)
		Confinement topical area	11	K. Burrell (GA)
		Boundary topical area	7	S. Allen (LLNL) P. West (GA)
		Heating and current drive topical area	6	R. Prater (GA)
		Topical area sum	28	
		Percentage of total days	47	
		Total allocated days	60	
		Contingency	15	
		Sum	75	
		Available days	75	

Topical Science areas. Each of the ten efforts has a <u>responsible leader</u> and deputy leaders. A brief synopsis of progress in the various thrusts in 2000 followed by year 2001 plans is given below.

4.1. RESEARCH THRUSTS FOR 2001

4.1.1. RESEARCH THRUST 1, H–MODE PEDESTAL AND ELMS (Leader: R.J. Groebner, Deputy: T.H. Osborne)

Thrust 1 seeks an understanding of the basic physics of pedestal formation and of energy losses due to ELMs. The research plan for 2001 will address the questions formulated in Section 2.4.1.

Two experiments will be conducted to study the physics that sets the width of the H–mode transport barrier. The goal of the first experiment, conducted in cooperation with the edge/divertor topical science group, is to determine if the shape of the edge electron density profile in H–mode is determined purely by neutral fueling. This experiment will test a model that relates the shape of the density profile to both the amount and location of neutral fueling. This model predicts that there is a relation between the width and height of the density barrier and this will be tested. In addition, measurements of the neutral deuterium density, which can now be made at the midplane and in the divertor, will be used to see if the width of the density barrier qualitatively tracks the penetration depth of the neutrals.

A second experiment will address this issue in a different way and seeks to determine if the barrier width is set purely by physics parameters or if atomic processes are also important. This test will be done by performing an edge similarity comparison with C–Mod. The execution of the experiment is to make plasmas that have pedestals whose non-dimensional parameters are identical to those of reference C–Mod discharges. The scale lengths in the density and temperature transport barriers will then be compared to see if they are the same. If so, the result would be evidence that purely physics parameters control the pedestal width; if not, the result would suggest that atomic physics, such as neutral fueling, plays a role in setting the pedestal width.

An experiment will be performed to study the mechanism by which ELMs couple to the core of the plasma. Specifically, the experiment will seek to determine if the eigenfunction of an ELM couples to the core via the q-profile. If so, it is expected that the penetration depth and energy loss of ELMs will substantially increase as lower order rational surfaces are introduced into the plasma. Thus, plasmas will be made to reduce the minimum q from above 2 to less than 1 in order to search for these effects.

In collaboration with Thrust 7, experiments will be run to determine what the parameters are that are required to produce the QH–mode. These will be determined by attempting to make the QH–mode over a wide range of parameters. Some of the important parameters include plasma current, toroidal magnetic field, plasma shape (such as triangularity) and density. Heating power will be an important variable for each discharge condition.

The structure of the EHO will be studied in an experiment that will attempt to first determine where the mode is located and then to determine what the mode is. There are two primary ideas about the location of the mode. One is that the mode is located at or very near the separatrix and the other is that it is a resistive tearing mode at the q=3 surface. The experiment will seek to distinguish between these ideas by using fluctuation diagnostics to search for a phase reversal of a fluctuation signal. The BES and reflectometer systems will be particularly important in this study. In addition, a slow variation of the edge q will be attempted during a discharge. It is expected that if the mode has an island-like structure, then a dramatic change in the signatures of the mode should be observed as the edge q is changed.

4.1.2. AT SCENARIO, RESEARCH THRUST 2 — PREPARATION FOR AN NCS AT PLASMA DEMONSTRATION (Leader: M. Wade, Deputies: T. Luce, J. Ferron)

The short-term (i.e., CY2000) research program within Scientific Research Thrust 2 will be focussed on addressing immediate physics issues that presently stand as obstacles to the achievement of the AT existence proof. These issues are: (1) can $\beta_N \sim 4$ be achieved in an optimized pumping configuration? and (2) can ECCD can be used effectively to modify/control the current density profile in a high performance plasma without having deleterious effects on other aspects of the developed scenario?

As discussed previously, attaining $\beta_N \sim 4$ and $\beta \sim 5\%$ in quasi-steady-state is a general requirement for fully non-inductive, high performance tokamak operation. Although this level of performance has been obtained in optimized magnetic configurations with $\kappa \sim 2.0$ and $\delta \sim 0.9$, experiments in 2000 showed that the β limit in plasma shapes compatible with adequate density control via divertor exhaust (with $\kappa \sim 1.8$ and $\delta \sim 0.65$) to be 10%–15% lower. Detailed follow-up experiments in 2000 showed that this β limit scaled roughly linearly with the shape parameter S = (I/aB) q_{95}. However, because the entire scan was done at fixed (I/aB), the dataset does not preclude a simple dependence on q_{95}. Experiments this year will seek to break this correlation by operating plasmas with the same shape (i.e., S = constant) but with different q_{95} by varying plasma current and toroidal field. Further studies will be carried out to investigate the role of plasma shape in achievable β_N with a view toward a possible future divertor modification. Also, the first use of resistive wall mode stabilization in these high performance plasmas will be made in hopes of gaining access to higher β_N states.

ECCD studies will focus on understanding current drive efficiency in plasmas operating near the marginal stability limit for both MHD and turbulence-driven transport. In particular, one would like to know if ECCD will have deleterious effects on MHD and transport. Although initial studies in this area are independent of the outcome of the search for higher β_N solutions, these studies will ultimately be affected by constraints imposed by such a solution. For example, if it's found that the observed β_N scaling is simply a q_{95} scaling, then to recover β_N ~ 3.8 in an optimized pumping configuration, the toroidal field will have to be increased to B_T ~ 2.1 T. The lower ECCD efficiency with the resonance further outboard will mean higher EC power will be required to achieve the scenario.

Upon successful resolution of these issues, it is envisioned that this research thrust will move aggressively towards integrating the essential ingredients (resistive wall mode stabilization, density control, and ECCD) into a combined scenario. Although we do not expect to have sufficient ECCD in the CY2001 to achieve a fully non-inductive existence proof, success at all steps of the plan should allow the demonstration of a plasma state that, although slowly evolving, has all of the essential ingredients of a steady-state, high performance advanced tokamak plasma.

4.1.3. RESEARCH THRUST 3 — VALIDATE NEOCLASSICAL TEARING MODEL AND INVESTIGATE STABILIZATION WITH ECCD (Leaders: R.J. La Haye, C.C. Petty) (4-1/2 days)

After the resistive wall mode instabilities that are the subject of Thrust 4, the next largest immediate stability concern for the AT work are the neoclassical tearing modes (NTMs). These modes have been seen to limit the performance in all our approaches to AT plasmas. Even in plasmas in which q_{min} has been raised above 2, NTMs (m/n=5/2) have been observed. The purpose of this thrust is to gain further physics understanding of the neoclassical tearing modes and continue to develop means of avoiding or stabilizing them.

This thrust has four highest priority tasks: (1) finishing the multi-device dimensionless scaling of the onset of m/n=2/1 NTMs, particularly compared to JET-EFDA, (2) finishing the suppression of m/n=3/2 NTMs by rad. loc. off-axis ECCD in presence of sawteeth instabilities, (3) doing a proof of principle of whether sub-resonant static externally applied helical field can suppress a rotating m/n=3/2 NTM and (4) trying ECCD suppression of an m/n=2/1 NTM in a q_{min} > 1 discharge.

Two principal research lines are foreseen in a three year plan: (1) studies in H–mode with sawteeth present and (2) studies in an AT mode with raised q_{min}.

H–Mode With Sawteeth

Work in 2001 will continue on our ongoing collaboration with JET, ASDEX Upgrade, JT–60U, and Alcator C–Mod on the scaling of NTMs. Particularly, the dependence of the m/n=2/1 critical beta dependence or dimensionless parameters ρ_{i*} and $(\nu_{ii}/\varepsilon)/\omega_{e*}$. Work will continue to follow up the successful complete suppression of a m/n=3/2 NTM by ECCD. This will include PCS real time position adjustment optimization. An alternate means of mode suppression will be given a proof of principle test; static sub-resonant helical fields (as can be produced with the C–coil such as m/n=1/3) of sufficient amplitude are predicted to reduce the rotating NTM helical pressure perturbation which sustains the mode.

AT–Mode Line

In 2001, we will make the first investigation of ECCD suppression of a m/n=2/1 NTM. This will be done in a discharge without sawteeth (but with $q_{min} \geq 1$) in which peak performance and/or duration is limited by the NTM.

In the year 2002, we expect to have developed the understanding and PCS control of NTM suppression to use it in a true $q_{min} \sim 2$ AT plasma, keeping non-resonant field suppression as an alternate if the PoP is successful.

Principal Goals for 2001

1. Finish study of ECCD suppression of m/n=3/2 NTM in presence of sawteeth instabilities.
2. Do PoP of non-resonant static helical field suppression.
3. Start study of ECCD suppression of m/n=2/1 NTM in discharges without sawteeth.

4.1.4. RESEARCH THRUST 4 — ESTABLISH LIMITS OF PASSIVE WALL STABILIZATION, BENCHMARK QUANTITATIVELY FEEDBACK MODELS AND OPTIMIZE CONTROL WITH IMPROVED SENSORS (Leader: A.M. Garofalo, Deputy: L.C. Johnson)

The AT Program on DIII–D has shown that the growth of *n*=1 resistive wall modes limits the maximum steady-state value of β_N to the $\beta_N^{no\ wall}$ stability boundary, even in presence of plasma rotation. Over the past two years, work in the DIII–D Research

Thrust #4 has demonstrated that feedback using the existing C–coil with external saddle loops as mode sensors can stabilize the RWM at β_N just above $\beta_N^{\mathrm{no\,wall}}$.

Guided by calculations performed using the three-dimensional feedback simulation code VALEN, new RWM sensors have been designed and installed in DIII–D, and will be available for feedback experiments in 2001. New saddle loops mounted inside the vessel are shorter and measure the radial field closer to the plasma than the external saddle loops. They are predicted to improve the stable beta limit up to 30% of the incremental gain achieved with an ideal wall.

Newly installed Mirnov probes increase to four the number of diametrically opposed measurements of the poloidal field inside the vessel. These sensors are predicted to increase RWM stability by 50% towards the ideal wall β_N limit.

Among the most important outstanding scientific questions that Thrust #4 intends to answer during the 2001 experimental campaign are:

- What are the limits of passive wall stabilization? Can we develop an optimized plasma regime able to maintain indefinitely the toroidal rotation?
- Do the new sensors improve feedback stabilization?
- Do new beta limiting phenomena intervene if we are successful at increasing the steady state beta value above the n=1 no-wall limit?
- Do experimental results on RWM stabilization agree with the VALEN predictions, to the extent that we can confidently use VALEN for designing an optimized feedback system (and feedback systems in future devices)?

The proposed experimental plan developed by the Research Thrust #4 to address these issues is shown below.

Experimental Plan for 2001

Wall Stabilization Physics: validate models of plasma rotation slowdown and RWM stability (2 days)

- Recent theory (Boozer, submitted to PRL) predicts the damping of toroidal rotation in a plasma near marginal stability. A systematic, controlled investigation of error field amplification will be conducted to determine the importance of error field correction in plasmas near or above the beta limit and to compare with theoretical predictions

- Systematic study of q-profile effects to compare with theoretical predictions of wall stabilization dependence on the number and location of resonant surfaces present in the plasma.

- Systematic study of plasma/wall separation to compare with theoretical predictions of wall stabilization dependence on the distance and location (inboard/outboard) of the wall.

RWM Feedback: demonstrate improved steady-state β_N limit through feedback stabilization of the RWM (7 days)

- Using SND target plasma with current ramp technique for reproducible RWM onset, test feedback system with new sensors and algorithms (internal saddle loops and internal, integrated Mirnov loops).

- Use feedback control of the neutral beam power to maintain a desired constant β_N in a steady state-like high performance AT target and establish β_N limit without RWM feedback.

- Using feedback control of the RWM amplitude, increase β_N in small steps from shot to shot, to test and measure the improvement in maximum stable beta.

- Compare performance of different RWM feedback algorithms (smart shell, mode control, fake rotating shell).

- Compare performance of different sensors (external saddle loops, internal saddle loops, internal, integrated Mirnov loops).

- Demonstrate real-time identification of RWM using multi-sensors, that is simultaneous use of toroidally and poloidally distributed Mirnov probes and midplane and off-midplane saddle loops inside the vessel.

- Identify the characteristic of new beta limiting phenomena that might intervene in the n=1 feedback stabilized steady state plasma above the n=1 no-wall limit (higher n external kinks and their RWMs, and NTMs).

- Compare experimental results with predictions of 3-D electromagnetics code VALEN and other feedback models.

4.1.5. THRUST 7 — INTERNAL TRANSPORT BARRIER CONTROL (Leader: E.J. Doyle; Deputy: C.M. Greenfield)

Goals and Background

The ultimate goal of Thrust 7 is to develop both the scientific basis for, and practical implementation of, transport control in the plasma core, so as to optimize AT pressure profiles for simultaneous high performance, improved stability and high bootstrap fraction. On a shorter three-year time scale, the goals are to develop profile control tools with a well understood scientific foundation so as to modify the spatial extent of internal transport barriers (ITBs), and explore potential ITB based high performance AT scenarios, such as the new QDB regime

As discussed previously in Section 1**,** the DIII–D AT physics program seeks to explore the ultimate potential of the tokamak. The Thrust 7 goals directly relate to exploring that ultimate potential in the following ways:

— Expanding the spatial extent of the ITB increases the plasma volume with suppressed anomalous transport, hence increasing fusion performance

— Broad pressure profiles with transport barriers near the plasma edge (ρ~0.7-0.8) are required in order to achieve large, well aligned bootstrap currents, and

— MHD stability modeling indicates that the maximum stable normalized beta increases with increasing ITB radius and half-width.

— In addition, increased understanding and control of core plasma transport will be required in order to successfully design and realize integrated, high performance AT plasmas.

Illustrated in Fig. 27 are the scientific issues and practical applications addressed by Thrust 7, along with their inter-relationships. The basis of the thrust is the scientific challenge posed by understanding and controlling core plasma transport. Such an understanding leads directly to general profile control applications. However, in higher performance plasmas integration issues become apparent – sustained transport control cannot be achieved without also considering MHD stability limits and q profile control. A compatible pressure profile, q profile and stability solution is required for sustainable high performance AT scenarios. Understanding and controlling plasma transport is an active and rich field of scientific inquiry. To illustrate this, the following transport control techniques are under development at DIII-D, all of which have a firm basis in theory and theory-based modeling: (1) Modification of the ExB shearing rate via toroidal momentum

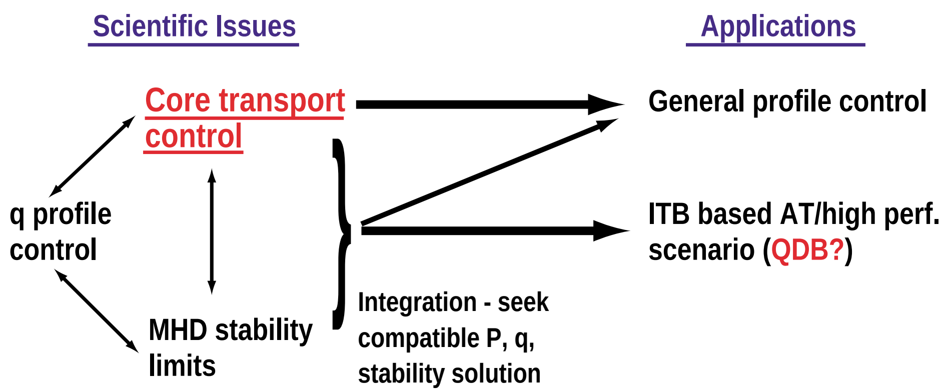

Fig. 27. Scientific issues and practical applications addressed by Thrust 7, and their inter-relationships.

injection by neutral beams; (2) modification of turbulence growth rates via changes in magnetic shear (q profile), impurity dilution, pellet injection or localized heating (e.g. ECH), and (3) turbulence stabilization via large Shafranov shift (α-stabilization).

Results Obtained in 2000

Thrust 7 controlled 4.5 run days in 2000. In addition, several related experiments were run in the Confinement and Transport Topical Science area. Highlights of results obtained in 2000 include the following:

- A new long-pulse, high performance operating regime, dubbed the Quiescent Double Barrier (QDB) regime, was discovered. The discovery of the QDB regime arose directly from theory-based work using counter-NBI discharges to expand the ITB radius, by modifying the interplay of terms in the expression for the E×B shearing rate. The QDB regime features a quiescent, ELM-free QH–mode edge, and the edge and core transport barriers are compatible. Although ELM-free, the QH–mode edge still provides density and impurity control, due to the presence of a continuous Edge Harmonic Oscillation (EHO). To date, the QDB regime has only been obtained in counter-injection discharges with divertor pumping. Compared to previous ITB discharges with L-mode edges, performance in QDB plasmas is higher because of the higher temperatures associated with the QH–mode edge.
- Impurity injection into pre-existing ITBs with an L–mode edge expanded the ITB radius and improved plasma performance. The physics of this improvement is understood in terms of a synergism of decreased turbulence growth rates and increased E×B shear.
- Modeling of electron thermal ITBs (eITBs) created with ECH indicates that the T_e gradient within the ITB is at marginal stability to the ETG mode, and that

α-*stabilization* effects are essential to eITB formation. (The experimental time for this work was carried out in the Confinement and Transport TS area).

These results are more fully described in the Thrust 7 DIII–D Year End Review document available on the web at http://fusion.gat.com/exp/2001/agenda.html as well as in papers from the 2000 IAEA and APS conferences.

Planned 2001 Experiments

The success of the 2000 experiments has accelerated consideration of both ITB based scenario development and integration issues. Here, integration refers to both the integration of multiple ITB control tools within a single discharge, and also the integration of transport, MHD stability and q profile issues. Shown in Fig. 28 is a flow chart illustrating the issues addressed by the planned six days of Thrust 7 experiments in 2001, and their inter-relationship. Investigation of the QH/QDB regimes will be shared with Thrust 1.

In more detail these experiments and issues are:

1. Investigate the physics, scaling and operational robustness of the new QDB regime. Should finish year with deeper physics understanding of the QDB and QH–mode regimes. (Two days Thrust 7, one day Thrust 1 allocation).

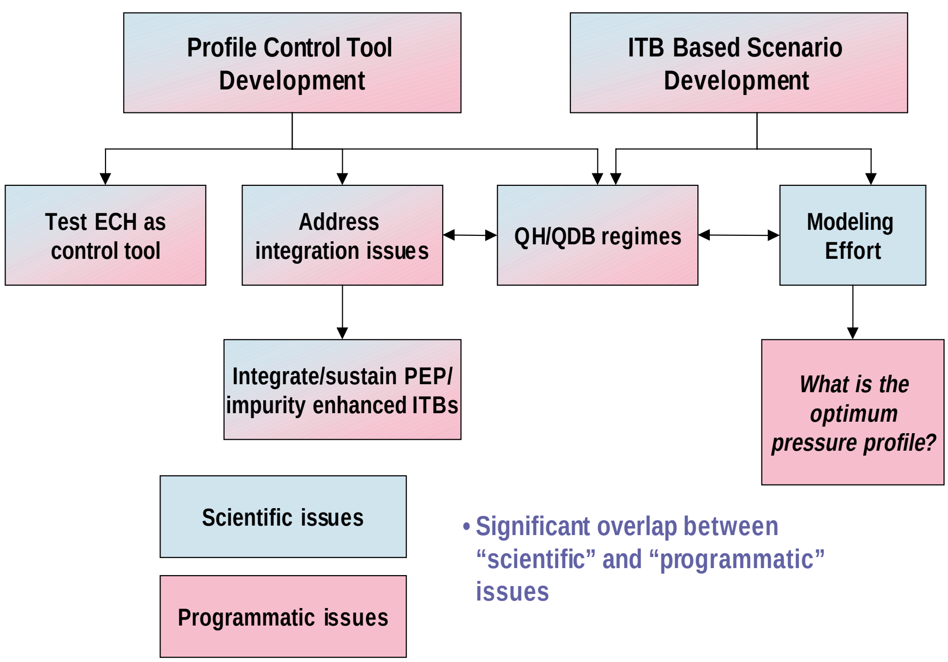

Fig. 28. Flow chart showing an outline of the issues addressed by planned Thrust 7 experiments in 2001.

2. Determine impurity transport in QDB plasmas and in particular assess role of density peaking. Is impurity transport and rotation neoclassical? Distinguish impurity source and transport effects. (One day).

3. Continue control tool development in QDB plasmas, which provide a long-pulse target plasma for integration of multiple control techniques and slow barrier expansion. Use control tools to decrease density peaking. (One day).

4. Initiate use of off-axis ECH as an ITB control tool. May provide precision control capability lacking in tools tested to date. (One day).

5. Integrate and sustain PEP and Impurity Enhanced ITBs. (One day).

6. Initiate modeling effort to determine optimal target pressure profile for future Thrust 7 efforts. In particular, assess the prospects for non-inductive sustainment of the QDB regime.

Issues for 2002 and Beyond

Depending upon both the experimental and modeling results obtain in 2001, there is a major decision point as to whether to continue to pursue the QDB regime as a potential high performance AT regime in 2002. Whatever the outcome of this decision, the QDB regime may still be useful in providing a long-pulse, quasi-steady state target plasma for profile control tool development. As further progress is made in benchmarking and understanding the various control tools available, emphasis on integration and "closed loop" profile control experiments will increase relative to the initial investigation of these issues planned for 2001. In addition, we expect the modeling effort initiated in 2001 to produce more concrete target pressure profiles, as well as guidance on what combination of our available control tools could be used to create and sustain these profiles. At all stages the experimental results will be used to both test and benchmark the transport analysis and modeling codes. By 2003, we anticipate having demonstrated and benchmarked a suite of profile control tools, and that the operation and understanding of each of these tools will rest on a firm scientific foundation.

4.2. PHYSICS TOPICAL AREAS

4.2.1. STABILITY (Leader: E.J. Strait)

In 2001, most of the stability experiments will be carried out under the research thrusts, and those plans will be described in more detail in the corresponding sections of this report. For example, neoclassical tearing mode stability experiments, which last year were part of the stability topical science area, will be carried out in 2001 under research Thrust 3. Here the primary goals include continuation of the control of instabilities with localized electron cyclotron current drive, and the first experimental test of the predicted stabilizing effect of non-resonant magnetic perturbations. Resistive wall mode stability experiments will be carried out under research Thrust 4. Here the primary goals are investigation of rotational stabilization physics including the possible role of error field amplification, and exploitation of a new set of internal magnetic diagnostics which are predicted to improve the feedback-controlled stability limits. Investigation of edge-driven instabilities is a strong element of research Thrusts 1 and 7, including the role of MHD stability in the newly discovered quiescent H–mode regime.

Experiments within the stability topical science area in 2001 will focus on validation of basic MHD stability physics, development of disruption mitigation, and exploration of new regimes. These experiments make use of DIII–D's extensive set of diagnostics for precise, detailed measurements of the pressure and current density profiles and the internal structure of MHD modes.

Disruptions are in principle predictable, occurring when a stability boundary is crossed, and much of the DIII–D stability program can be viewed as learning how to predict and avoid disruptions. However, some disruptions are inevitable due to unforeseen causes such as control system failure or unexpected impurity influx. The experiment planned for this year continues to develop methods of mitigating the effects of disruptions using strong gas puffing. Important physics issues to be investigated include the transport of impurity ions and neutral atoms, physics of non-axisymmetric halo currents, and the role of avalanche processes in runaway electron generation.

Validation of resistive interchange mode theory in regions of negative magnetic shear will provide a test of basic stability physics that is also important to the stability of stellarator plasmas. Resistive interchange modes have previously been observed in DIII–D discharges in a central region of negative magnetic shear. The planned experiment will obtain detailed measurements of the internal mode structure for comparison with theoretical predictions, using the full range of fluctuation diagnostics, and make a systematic study of the dependence of the stability threshold on the local pressure gradient and magnetic shear, again for comparison to theory.

Investigation of the physics of the sawtooth crash will be continued, including the role of the Mercier stability criterion. The primary goal is to compare a bean-shaped plasma that remains Mercier stable in the core because of the strong shaping, and a plasma with an elliptical cross-section that is Mercier unstable at the core. The expected result is a Kadomtsev-type full reconnection at the sawtooth crash in the first case, and short-wavelength instabilities and only a partial reconnection in the second case. If time permits, the central current density profile will be modified using ECCD.

The stability properties of discharges with very low global magnetic shear will be explored. This represents a possible new operating regime for high fusion performance. It is well known that both the energy confinement and the beta limit increase with plasma current. However, many previous experiments have found a strong degradation of confinement relative to this scaling as the safety factor q_{95} decreases below about 4. On the other hand, more recent experiments with negative central shear showed no degradation down to $q_{95} \sim 3$, perhaps because of the absence of sawteeth. The planned experiment will explore the possibility of high absolute beta and energy confinement in configurations with $q_{95} < 3$ and $q(0) > 1$; if the central q can be increased above 2, there is also the possibility of a significant fraction of bootstrap current.

4.2.2. CONFINEMENT AND TRANSPORT — 11 DAYS (Leader: K.H. Burrell)

This topical area has experiments under various working group headings: in the area of fundamental turbulence studies, we will perform two experiments. In the first, we will change the ratio of poloidal to toroidal field while holding all other nondimensional paramters fixed in order to investigate whether the radial correlation length of the plasma turbulence scales with the poloidal or toroidal gyroradius. Different theories contain different predictions; this experiment is designed to test which of these are correct. The second experiment will be a first attempt to directly measure the properties of the zonal flows. These are stable, poloidally and toroidally symmetric perturbations which couple to the unstable turbulent modes and regulate the turbulence by extracting energy from it. There are possbile techniques with beam emission spectroscopy in the plasma core and with Langmuir probes at the plasma edge which may allow us to directly measure the zonal flows; these will be explored in this experiment. This experiment will also obtain information on local transport scaling with safety factor q; this latter experiment was originally proposed in the nondimensional transport area.

In the H–mode physics area, two days of experiments are planned. First, we will take half a day to investigate the effects of scrape-off layer flows on the power threshold for the L to H transition. There are theoretical predictions that these flows can have an effect and we want to test these. On the second half day of this experiment, we will make further investigations of the physics of pellet-triggered L to H transitions. The emphasis in this year's experiment will be on detailed, high time resolution measurements of the changes in the edge conditions across the pellet-triggered transition. On the second experimental day,we will investigate further the effect of the ion ∇B drift on the edge plasma prior to the transition. We wish to extend last years observation of a turbulence velocity shear layer to a wider parameter range. We especially want to study the shear layer for the case with unfavorable ∇B drift for heating powers just below the threshold. Last year's work was done with the same heating power for both directions of the ∇B drift, which meant that the input power was well below threshold for the case with ∇B drift away from the X-point. We expect the shear layer to develop in both cases when the input power is close to threshold power; the experiment will test this expectation.

In the area of test of transport models, an experiment will be done to demonstrate the existence of a heat pinch with outside launch, second-harmonic ECH and to determine if the heat pinch is dependent on the sign of the magnetic shear as predicted. The inward transport effect seen with the 60 GHz system remains a sever challenge to the theoretical community. One remaining mechanism could explain the observed profiles without requiring transport up the temperature gradient: the conversion of the fraction of ECH power which is not absorbed at the resonance to electron Bernstein waves at the upper hybrid layer. This mode conversion is not possible with second harmonic outside launch. The superior diagnostic set now available and the higher power densities possible with the 110 GHz ECH system could provide clear evidence of the mechanism responsible for the inward transport. Furthermore, the theoretical heat pinch model of coupled transport between ∇-J and ∇-T can be tested by comparing the non-diffusive electron transport for positive and negative shear plasmas.

A second test of transport models experiment is planned to provide tests of turbulence simulations, tests of transport models with modulated ECH, a test of the predictive capability of turbulent transport models and a demonstration of marginal stability in the electrons (L–mode part only).

Two days of experiments are planning in the nondimensional transport area The first will complete the experiment on the elongation scaling of transport which was begun last year. As a part of this work, we will also investigate the effect of Shafranov shift stabilization of turbulence. This latter experiment was originally proposed in the core transport physics area but it combines almost perfectly with this elongation experiment. The second experiment in the nondimensional transport area will investigate the changes in radial correlation length and decorrelation time as the electron to ion temperature ratio is varied. The analogous experiment last year in this area investigated the same changes with normalized gyroradius.

The core transport physics area will have three days of experiments. First, we have decided to increase the empahsis on angular momentum transport work this year because an ability to predict the toroidal rotation is central to our ability to predict the E×B shear which is important for turbulence reduction. The issue to be confronted in this experiment is to test whether the difference in toroidal rotation amoung the various ion species in the plasma agrees with the predictions of neoclassical theory. If it does, then we can use the theory to determine the main ion toroidal rotation from the measurements of impurity rotation. A variety of plasma conditions will be used here to test the theory as completely as possible. In addition, we will use measurements taken in Thrust 7 to expand this parameter range to counter injection.

Second, we will test our understanding of turbulence stabilization at various length scales by attempting to create plasmas with simultaneous electron and ion core transport barriers. Theory suggests that E×B shear stabilization is useful for stabilizing fairly long wavelength turbulence such as that seen with ion temperature gradient modes. Shafranov shift or alpha stabilization is predicted to be effective in stabilizing electron temperature gradient modes. Experiments last year created electron transport barriers with localized ECH. We have a long history of making transport barriers with neutral beam heating; in the best shots, we have transport barriers in all four transport channels. The goal this year will be to utilize combined heating to see if our theortical understanding is correct.

Third, we will perform a series of experiments to make detailed tests of electron thermal transport. Most of these will utilize modulated ECH in various fashions to probe the electron transport both inside and outside the core transpor barrier.

4.2.3. EDGE AND DIVERTOR PHYSICS (Leader: S.L. Allen)

We have six experiments planned for the upcoming 2001 campaign, listed below with a brief description.

1. Measurements of turbulence and the effects of drifts in near double-null operation. Measure SOL turbulence and heat and particle flux profiles and asymmetries as dRsep is varied about a double-null configuration (–2 cm < dRsep < +2 cm). Compare with BOUT and UEDGE. (Approximately 1 day led by Tom Petrie, GA).

2. Far SOL transport and recycling. measure non-diffusive transport mechanisms in the far SOL and main chamber recycling and fueling in L&H mode LSN plasmas, compare with C-Mod results and SOL transport theories. (Approximately 1 day led by Dennis Whyte, UCSD).

3. Measurement of potentials and drifts in single-null operation. Document SOL, X-point, and divertor plasma potentials, pressures, and drifts in USN and LSN and in forward and reverse B_T across an L to H transition. Compare to UEDGE. (Approximately 2 days led by Mike Schaffer, GA).

4. Experimental documentation of a simple plasma and exposure of a DiMES Li sample. Document an L–mode plasma (no ELMs) using our spectroscopic, visible

and UV TV, IRTV, fixed and plunging probes, IRTV, and bolometric diagnostics for detailed comparison to UEDGE and onion skin models. Expose a DiMES lithium sample to this weel documented and modeled plasma. (0.5 day, led by Dennis Whyte, UCSD).

5. Test of pedestal density theories and postulates. Investigate the role of fueling efficiency and particle loss in the achievement of high pedestal density. Compare to theoretical predictions and determine a path to high density at high confinement. (Approximately 1.5 days led by M.A. Mahdavi).

6. Carbon sources, flat vs contoured tiles. Measure the local carbon source and the core carbon contamination as the inner strike point is moved across the division on the inner wall between the new, well engineered, contoured tiles and the old flat tiles. Also measure carbon penetration from methane puffing at several poloidal locations. Compare to UEDGE and Monte Carlo models. (Approximately 1 day led by C. Lasnier).

4.2.4. HEATING AND CURRENT DRIVE PHYSICS (Leader: R. Prater)

The three year goals of this topical science area are described in Section 3.2 of this report. The two key goals of highest priority for the 2001 campaign are development of a predictive model for ECCD and development of discharges with high bootstrap fraction.

For ECH physics, three experiment-days are planned. The key objectives of the experiments are:

1, Complete the ECCD scans (as described in Section 3.4) from last year. The analysis of those scans will surely provide some surprises and suggest some new studies which will be necessary for development of a complete model of ECCD. At the least, some data points will need to be revisited to validate previous results.

2, Extend the previous scans of ECCD to larger minor radius. This goal is motivated by the fact that ECCD is needed by the AT program at minor radii around 0.5 to 0.7. We don’t yet have direct measurements of ECCD at such large minor radii.

3, Determine the dependence of ECCD on electron beta. It is believed from theory models that ECCD at high electron beta will reduce the deleterious effects of trapping, resulting in strongly improved current drive efficiency even for large minor radius. This concept needs to be tested since it is key for the AT applications. The electron beta will be increased by adding heating power (ECH plus NBI) and by operating in the H–mode.

The first objective can be carried out using two gyrotrons, since only two antennas (in the P1999 launcher) have the flexibility to perform such scans. The second and third objectives will require higher power from three or four gyrotrons operating with nearly the design power.

For bootstrap studies three days are planned. The objectives include:

1. Develop discharges with high bootstrap fraction (2 days). A steady-state reactor will have bootstrap fraction near unity. This links the profiles of pressure and current in a new way which we need to understand. The urgency is that high bootstrap fraction is one leg of the DIII–D AT program (high beta, high confinement, and high bootstrap fraction), and it has not been addressed in the present fusion program. The approach is to use low current discharges with high density (to minimize the NBCD and the flux diffusion time) and to maximize the ratio of rf heating to NB heating.

2, Measure the dependence of the edge bootstrap current on collisionality and shape. The urgency is that the edge bootstrap current is known to have a strong effect on plasma stability, but the models for edge bootstrap have not been validated. The edge bootstrap current may be directly measurable by operating the discharge without current feedback in order to reduce the effect of ohmic flux on the edge current. The concept is to measure edge bootstrap current for a range of triangularity and collisionality and compare with calculations.

4.3. THE 2001 OPERATIONS SCHEDULE

The operations schedule is designed for efficient and safe use of the DIII–D facility. Seventeen calendar weeks of plasma physics operations is scheduled for the calendar year 2001. The plan is to have four 4- or 5-week run periods. The operations schedule is shown in Fig. 29. Operations are carried out on either 4 or 5 days per week for 8.5 hours. Six Thursdays, where there is no operation scheduled for the following day, operation is extended to 10.5 hours to allow longer experiments to reach completion or do some very short experiments.

In addition to operating the tokamak, maintenance has to be performed and new hardware is being installed to enhance DIII–D capabilities. The schedule for these activities is for the maintenance to be done when the tokamak is not operating.

Fig. 29. Operations Schedule.

The plan takes into consideration factors such as efficient matching of the machine run time with the availability of hardware and data analysis capabilities. Above all, the DIII–D program is carried out to keep radiation exposure to employees and to the general public as low as reasonably achievable (ALARA) and still carry out the research program.

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